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5MW Low Power Reactor

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5MW Low Power Reactor

Preface

926B0125A Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 p 1

[Article by Ren He [0117 0735]: "Preface"]

[Text] The 5MW low power reactor (5MW LPR) was produced by rebuilding the former uncompleted low power reactor (LPR). The design power of the LPR was 100 kW and the power of the reactor (5MW LPR) after rebuilding is 5MW.

The design objective for the LPR was to study the physical performance of the reactor core under HFETR fueled working conditions. The objective of rebuilding was, with a prerequisite of maintaining the original design functions, to increase the reactor power and to add and perfect the facilities and equipment in all of its systems to enable monocrystalline silicon neutron irradiation doping, isotope production, irradiation to recolor gemstones, and other production work and to provide services to neutron activation analysis and other research work. This research was done mainly using fuel elements unloaded from the HFETR (with a mean specific burnup no greater than 40 percent) to do research on deeper burnup (no more than 45 percent) under conscientious monitoring. During the operation process, if it could be fully confirmed that no damage occurred to the fuel elements, analysis and evaluation would provide a reliable foundation for safer development and utilization. This is a new type of practice in fuel research and has great economic significance. This would make the 5MW LPR a dual-use reactor for doing various types of production while also taking scientific research into consideration.

The guiding principles for rebuilding the LPR were: 1) Make every effort to utilize the original design, try to conserve, and make full use of the related experimental data, experience, and related conditions from the HFETR; 2) The designs of the overall unit and each of the systems had to be safe, reliable, and appropriate.

Work to rebuild the LPR got underway in October 1986. It received a fuel loading permit from the National Nuclear Safety Administration in January 1991 and completed a 72-hour full-power operation test on 2 August 1991, which formally completed the 5MW LPR. Completion of this reactor is the result of the support of all relevant leading departments of higher authorities and the major efforts at cooperation and joint striving of all related units in the academy. The cost of rebuilding this reactor was only 4 million-plus yuan and it is another accomplishment of our academy to foster the spirit of large-scale cooperation, rely on our own efforts, and arduous struggle. The completion of this reactor further strengthens our academy's new product development capabilities and gives it a reactor engineering research tool. Continual development and utilization of this reactor will make it play a more active role in spurring development of China's nuclear industry.

5MW Low Power Reactor: Design Parameters

926B0125B Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 pp 2-6, 17

[Article by Tang Xueren [0781 1331 0088], Lu Guangquan [0712 0342 0356], Lin Zhang [2651 3864], Zhao Zengqiao [6392 1073 0294], Shi Jingxian [0168 2529 6343], Sun Maoyu [1327 5399 3768], and Gao Xingdou [7559 2502 2435] of the China Nuclear Power Research and Design Academy, Chengdu: "The 5MW Low Power Reactor (5MW LPR)"; manuscript received 15 February 1992, revised manuscript received 18 March 1992]

[Text] Abstract:This article describes the primary design parameters, facilities, and characteristics of the 5MW LPR. The main characteristics of this reactor are its use of fuel elements unloaded from the HFETR and installation of a damage monitoring system to monitor each of the elements inside the reactor. It can be used for monocrystalline silicon neutron irradiation doping, molybdenum-technetium isotope production, irradiation coloring of gemstones, and so on.

Key terms: 5MW low power reactor, unloaded fuel elements, element damage monitoring, irradiation, monocrystalline silicon, molybdenum-technetium isotopes.

I. Introduction

The 5MW LPR was made by rebuilding the former low power reactor (LPR). The former LPR was a matching facility for the HFETR and is a "swimming pool" type reactor that uses fuel elements unloaded from the HFETR. The goal in building the reactor was to study the physical performance of high flux reactor fuel elements under fueled conditions, and its original design power was 100 kW.

Based on the original design, the plant building, "swimming pool", and so on had already been completed and part of the facilities were installed. Because the power was very low, no loop system, thermotechnical measurement system, element damage monitoring system, dose system, and so on were installed. Moreover, the power supply system, ventilation system, special drainage system, and shielding capabilities were only suitable for the 100 kW power scale. During the past few years, the halted state of reactor construction led to capital overstocks and idle equipment and instruments.

To change this situation, make comprehensive use of the LPR, and undertake monocrystalline silicon neutron irradiation doping, molybdenum-technetium isotope production, irradiation color alteration of gemstones, neutron activation analysis, and other research work, and in particular research on deeper mean burnup (no greater than 45 percent) for the fuel elements used in the HFETR with a specific burnup of no greater than 40 percent, design work to rebuild the LPR got underway at the end of 1984. With approval from the State Economic Commission, the rebuilding work began in October

1986. Equipment installation and single system debugging was completed in August 1990 and the final safety analysis report was completed and passed examination and acceptance in November 1990. Non-nuclear debugging was completed in December 1990 and it received a first heat fuel loading permit from the National Nuclear Safety Administration on 31 January 1991. The first criticality was attained on 3 February 1991 and a 72-hour full-load continuous operation test was completed on 2 August 1991, completing the 5MW LPR.

II. General Description of the Reactor

A. Overall configuration and primary design parameters

The overall configuration of the 5MW LPR is illustrated in Figure 1. The reactor core is immersed in a 8.5 m long, 2.2 m wide, 8.98 m deep stainless steel-lined pool. A lower support tube is installed at the bottom of the core and the lower support tube is connected with the reactor outlet cooling water mother pipe. The cooling water passes from the top of the reactor through the core and lower support tube and then to the reactor outlet mother pipe and heat exchangers. After cooling, it is returned to the pool. The control rod guide tubes installed in the core extend to the top of the core. Ion chamber guide tubes are installed around the activated region and the ion chambers and fission chambers are installed inside the guide tubes. Their measurement cables run upward from the guide tubes into the control room. The element damage detection monitoring tubes extend from the lower part of the lattice plate into the measurement space. Because the reactor power was increased 50-fold, a primary water cooling system, purification system, auxiliary cooling system, element damage detection system, secondary water cooling system, thermotechnical measurement system, dose system, and reliable power supply system were added. Moreover, the corresponding measures were adopted to increase the shielding capabilities of the reactor building and technical workshop, the control system was completely replaced, and the corresponding upgrading was done on the power supply, ventilation, water replenishment, special drainage, and other systems.

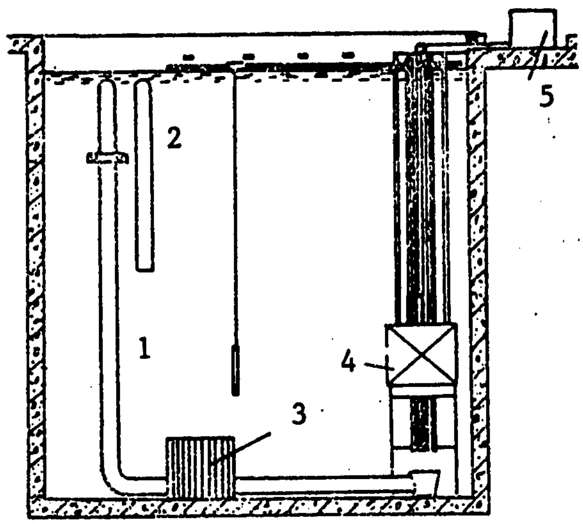


Figure 1. Overall Configuration of the LPR

Key: 1. Reactor outlet mother pipe; 2. Reactor inlet mother pipe; 3. Element storage rack; 4. Core; 5. Drive mechanism

The primary design parameters for the 5MW LPR are listed in Table 1.

Table 1. Primary Design Parameters for 5MW LPR

Item	Value
Reactor power, MW	5
Number of fuel element boxes loaded into core	32
Maximum thermal neutron flux (E_n<0.625 MeV), h/cm^2 x s	8.03 x 10^{13}
Maximum fast neutron flux (E_n≥0.625 MeV), n/cm^2 x s	1.43 x 10^{14}
Maximum heat flow density of fuel element surfaces, kW/m^2	269
Highest wall temperature of fuel element surfaces, °C	108
Water flow speed inside fuel element boxes, m/s	1.5
Fuel replacement schedule, MW x d	300
Moderator temperature coefficient, (ΔK/K)/°C	-16. x 10^{-4}
Minimum burnup ratio	6.6
Primary cooling water flow rate, t/h	620
Primary cooling water inlet temperature, °C	40
Primary cooling water outlet temperature, °C	47
Purification system flow rate, t/h	6.2
Auxiliary cooling system flow rate, t/h	54
Secondary cooling water flow rate, t/h	1,000

B. Reactor body

The core of the reactor body is configured with 32 boxes of elements, 44 beryllium blocks used to form a reflecting layer, 99 aluminum fill blocks, and 48 stainless steel fill blocks. There are four φ180 and four φ120 monocrystalline silicon irradiation ducts installed around the beryllium reflecting layer, and there is one φ63 molybdenum-technetium isotope irradiation duct at the K_{11} position in the center of the active region. All of these irradiation ducts are placed on a 235 mm thick stainless steel lattice plate, and the lattice plate is 1.4 m in diameter. The lattice plate is connected to the bottom surface of the pool by the upper and lower support tube bases. An activation analysis loop is placed inside the reactor and gemstone and other irradiation ducts can be installed as needed. Two of the 10 control rod guide tubes in the reactor core are used as safety rods, two are used as automatic regulation rods, and six are used as compensation rods. The absorbers in the control rods are identical to those in the HFETR and all of the followers are small beryllium blocks. Around the core there is a polygonous stainless steel inner barrel encircling the core and to the outside there is also a stainless steel outer barrel. The two layers of barrels also play a role in anchoring the core and thermal shield. The configuration of the core is illustrated in Figure 2.

The 5MW LPR uses fuel elements unloaded from the HFETR with a mean specific burnup no greater than 40 percent[1]. These elements are composed of six layers of concentric fuel tubes and inner and outer guide tubes. The fuel tubes were formed by co-extrusion. The fuel tubes are 1,100 mm long and have fuel sections 1,000 mm long. The fuel bodies are uranium-aluminum alloy and the u-Al_4 was used as a disperse phase and dispersed in the aluminum matrix. In them, the ^{235}U enrichment is 90 percent. This type of fuel has advantages like good irradiation stability, high heat conductivity, does not easily react chemically with the aluminum cladding and water, has good processing properties, and so on.

The deeper burnup experiments in the HFETR indicated that safely using elements unloaded from the HFETR with a mean burnup no greater than 40 percent only requires that the mean burnup not exceed 45 percent.

C. Loop systems

The key content of rebuilding the LPR was improving its cooling capabilities to increase the reactor power. For this reason, reactor loop systems were installed. They include a primary water main cooling system, purification system, auxiliary cooling system, and secondary water system.

1. Primary water main cooling system The main equipment in the primary water main coolant system are two main pumps (their electric motor power sources are connected by two independent circuits to external power sources for their power supply) and two heat exchangers. After the primary water enters the reactor pool through the reactor mother pipes, it flows via two branch loops in the primary loop, each of which enters a main pump and heat exchanger, and then it is collected into the mother pipes and returned to the reactor pool. The system uses a single element control configuration with one main pump and one heat exchanger forming an element. This type of configuration is simple to operate and enabled use of an upgraded original hoist building. The electric motors of the main pumps are fitted with flywheels to increase the inertial flow rate. The heat exchangers utilize a fixed tubesheet tube matrix structure. In the heat exchangers, the pressure of the secondary water is greater than the pressure of the primary water, so ruptures and leaks in the tubes in the heat exchangers cannot contaminate the environment.

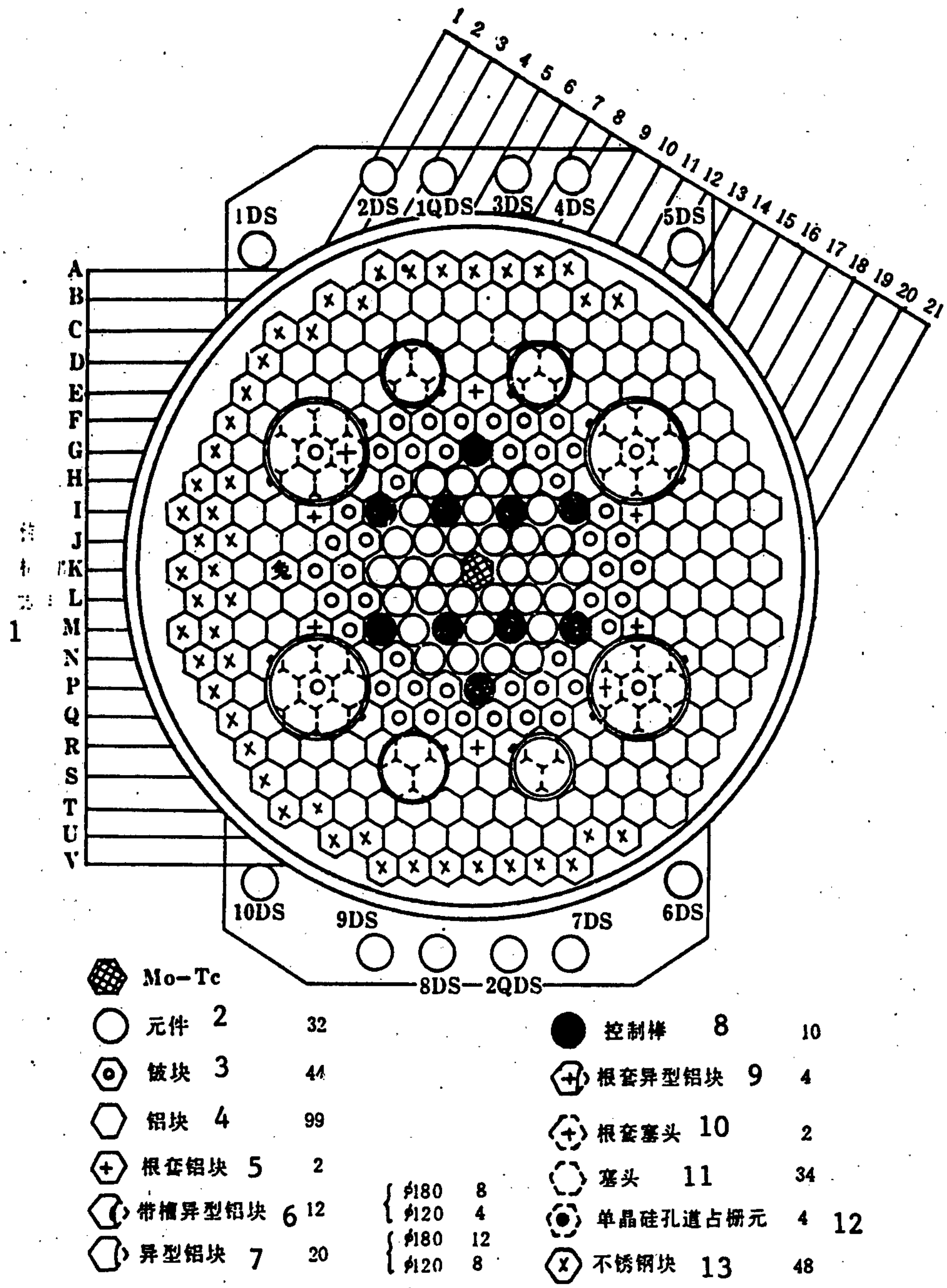
2. Purification system The purification system is mainly composed of a purification pump, front mechanical filter, two mixing beds, and rear mechanical filter. The system operation modes are relatively flexible. It can operate independently and it can operate simultaneously with the main systems. The volume ratio of the cathode and anode resins in the mixing bed is 1:2. The purification flow rate is 1 percent of the primary water flow rate, 6.2 t/h. The purification system extracts water downstream from the heat exchangers in the primary water cooling system, which eliminates the need for a purification cooler.

3. Auxiliary cooling system The auxiliary cooling system was designed to remove excess heat from the reactor. The auxiliary cooling system is mainly composed of two parallel-connected pumps (one pump operates, the other pump is used as a reserve) and the associated piping and valves. There is an interlock between the auxiliary cooling pumps and the main pumps and an interlock between the two auxiliary cooling pumps. When there is a simultaneous loss of external power to the two loops and the main pumps cease operation, one of the auxiliary cooling pumps goes into operation automatically. Debugging tests indicate that it takes no more than 6 seconds for the auxiliary cooling pumps to begin operation and attain their rated flow rates. The rated flow rate of the auxiliary cooling system is 54 t/h, which is sufficient to remove the residual heat released from the reactor.

4. Secondary water cooling system The feedwater in the secondary water cooling system comes from the HFETR feedwater pipes into the 5MW LPR secondary water operation room and is divided into two loops, each of which passes through a heat exchanger and then collects into the primary loop for discharge. There are three secondary cooling water pumps. During normal operation, one pump is held in reserve. The rated flow rate of the secondary cooling water is 1,000 t/h. The secondary cooling water passes through contaminated water monitoring prior to being discharged.

D. Element damage detection and positioning system

The element damage detection and positioning system was installed to monitor damage to the elements. It is



1QDS and 2QDS are startup ionization chambers used for cycle protection;
2DS and 7DS are ionization chambers used for power regulation;
1DS, 5DS and 8DS are ionization chambers used for power protection;
3DS, 6DS and 10DS are for reserve;
4DS and 9DS are ionization chambers used for power measurement.

Figure 2. Configuration of Core

Key: 1. Drive mechanism direction; 2. Elements; 3. Beryllium blocks; 4. Aluminum blocks; 5. Base collar aluminum blocks; 6. Grooved special aluminum blocks; 7. Special aluminum blocks; 8. Control rods; 9. Base collar special aluminum blocks; 10. Base collar plugs; 11. Plugs; 12. Lattice elements taken up by monocrystalline silicon ducts; 13. Stainless steel blocks

composed of two damage detection pumps (one pump operates, the other is held in reserve), a delay water tank, a delayed neutron monitoring station, a total γ monitoring station, sampling tubes, operations valves, and measurement instruments. One sampling tube is installed at the outlet of each of the 32 element boxes inside the reactor to divide them into five groups. A sampling tube is also installed on the primary water reactor outlet mother pipe. All of the sampling tubes enter the damage detection and monitoring room. The damage detection pumps extract water samples which pass through the delay water tank. It takes about 80 seconds for the water samples to reach the delayed neutron monitoring station. Delayed neutron and total γ measurements are made of the water samples to determine whether or not damage has occurred. During normal reactor operation, only the reactor outlet mother pipe is measured. If abnormalities are discovered, the valves can be switched over to monitor any group or any box of elements. The delayed neutron monitoring instruments and total γ monitoring instruments are both located in the main control room. When the delayed neutron or total γ count exceeds the specified values, an alarm signal is automatically issued.

E. Control and protection systems

The control and protection systems for the 5MW LPR use a one-of-two sampling combined logic program. The rod control system contains six sets of compensation rod drives and logic devices, two sets of power regulation systems, and two sets of safety rod drive systems. It takes less than 1 second for the safety rods to drop to the bottom. The nuclear measurement instruments include nuclear power measurement devices, broad-range cycle monitoring devices, nuclear power protection devices, and a fixed value amplifier used for power regulation.

This reactor has 15 accident signals and 41 warning signals.

The thermotechnical hydraulics parameters measured include temperature, pressure, flow rates, water levels, reactor thermal power, and so on, and there are a total of 44 measurement points. The important detection parameters include reactor outlet water temperature, hot box element outlet water temperature, primary cooling water flow rate, reactor pool water level, reactor thermal power, and so on.

F. Other systems

The 5MW LPR also has a power supply system, ventilation system, water replenishment system, special drainage system, technical transport system, fire control system, and communications and broadcast system.

III. Characteristics of the 5MW LPR

A. Uses elements unloaded from the HFETR

The 5MW LPR uses elements unloaded from the HFETR. The design mean specific burnup of the HFETR elements is 36 percent and they have been operated to a burnup of about 40 percent in the HFETR and are now being utilized in the 5MW LPR and operated to a mean specific burnup that does not exceed 45 percent. The elements unloaded from the reactor must undergo deeper burnup and this is carried out in another reactor. There is no previous example of this in China. To achieve this objective, transportation problems had to be resolved. Because the fuel elements are extremely expensive and "delicate" and the hot elements had very intense radioactivity, transporting them is extremely difficult. Based on careful element transport procedures, a transport program using a forklift, lead cask, and underwater loading and unloading was adopted for successful transport of the 32 boxes of elements required for the first heat for the 5MW LPR.

B. Has a positioning and damage monitoring system inside the reactor

The damage monitoring system for the 5MW LPR directly monitors damage to the 32 boxes of elements in the reactor without having to remove the elements from the reactor. Using this system permits relatively rapid discovery of early damage to the elements, avoids expansion of accidents, and guarantees safe reactor operation.

C. Protection system has in-service inspection functions

The protection system for the 5MW LPR does not use a two-of-three or two-of- four sampling configuration like nuclear power plants usually adopt. Instead, it has one-of-two sampling configuration safety logic device. While this might increase the frequency of reactor shutdowns, it does not reduce safety and is more economical. The two sets of safety logic devices both have in-service inspection functions with a detection cycle of 7 seconds that can satisfy safe reactor operation monitoring requirements.

D. Makes full use of irradiation space

One objective in rebuilding the LPR was to use it to irradiate monocrystalline silicon. To meet the demand in China and foreign countries for irradiation of large-diameter monocrystalline silicon, large diameter irradiation ducts were designed for this reactor and the monocrystalline silicon to be irradiated can have a maximum diameter of up to 15.2 cm. This reactor also has a molybdenum-technetium irradiation duct, activation analysis duct, gemstone irradiation duct, and so on. For this reason, it can make full use of the irradiation space and perform more functions.

E. Excellent economy

The principal aims in rebuilding the LPR were: 1) To make every effort to utilize the existing facility and equipment in storage, make full use of the relevant experimental results and experience from utilization of the HFETR, and, with a prerequisite of satisfying rebuilding performance requirements, to strive to conserve expenditures; 2) The design of the overall unit and

each of the systems must all be safe, reliable, and useable. During the rebuilding process, adherence to these principal aims enabled the successful rebuilding of an unfinished reactor with a design power of 100 kW into a reactor with a power of 5MW that can be used both for irradiation production and experimental research at a cost of only 4 million-plus yuan renminbi.

IV. Conclusion

Completion of the 5MW LPR has opened up a new path to full utilization and deeper burnup of elements unloaded from the HFETR, and it provides excellent conditions for research on large-dimension monocrystalline silicon neutron irradiation doping, molybdenum-technetium isotope production, gemstone irradiation coloring, and so on. This is another of China's accomplishments in low power reactor development and construction. However, because of restrictions by capital and other conditions during the construction process, construction of the secondary water discharge system and other things require further improvement. With continual development and perfection, the 5MW LPR will play a growing role in the development of China's nuclear industry and nuclear technology.

Reference

[1] Wang Jiafeng [3769 1367 0023], He Dongli Gongcheng [Nuclear Power Engineering], Vol 2, No 3, 1981 p 25.

Body Structure Design for 5MW Low Power Reactor

926B0125C Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 pp 18-22

[Article by Hu Guozhong [5170 0948 6850], Sun Linzhi [1327 2651 1807], and Liu Mouzhou [2692 6180 3166] of the China Nuclear Power Research and Design Academy, Chengdu: "5MW Low Power Reactor Body Structure Design"; manuscript received 20 February 1992]

[Text] Abstract: This article introduces the guiding ideology, materials selection, technical parameters, core configuration, and structural characteristics of the body structure design of the 5MW low power reactor (5MW LPR). The reactor body is the core part of the reactor. Its design content includes the structure and configuration of the fuel elements, control rods, reflecting layers, irradiation tubes, shielding, and nuclear measurement devices, the shape of the core supports, assembly of the components inside the reactor, and so on.

Key terms: 5MW low power reactor, reactor body, design principles, structural characteristics.

I. Introduction

The 5MW LPR is a matching project for the high-flux engineering test reactor. Its original design power was 0.1MW and it had swimming pool-type natural convection cooling. The core configuration is basically identical to the high-flux reactor and is arranged in equilateral triangle regular rows based on a 64 mm lattice distance. The central part contains the fuel elements and control rods and the exterior is the beryllium and aluminum reflecting layers. To expand the uses of the low power reactor, the power was increased to 5MW, which changed the reactor parameters and operation mode and changed from natural convection cooling to dual-loop forced cooling. To adapt to the physical, thermotechnical, shielding, measurement, and utilization requirements, the reactor body structure had to be rebuilt on its original foundation.

II. Design Principles and Main Design Parameters

A. Design principles

1. Safety and reliability. The operational safety of a reactor is the first question to consider. The design must satisfy the relevant standards, regulations, and quality assurance procedure requirements. When reactor safety is guaranteed, appropriate consideration can be given to its advanced properties.

2. Economy. Because this is a rebuilt reactor, every effort was made to use the original equipment and plant building to do more with less money.

3. Simple structure, convenient operation, adapted to the need for frequent replacement of irradiated products.

4. Substantial flexibility in core configuration to satisfy multi-objective irradiation tasks.

5. Every possible effort to use the design experience and experimental data from the high-flux reactor.

B. Main design parameters

Working pressure:	Normal pressure
Liquid elevation static pressure:	0.083 MPa
Design pressure:	≈ Normal pressure
Reactor inlet coolant temperature:	40°C
Reactor outlet coolant temperature:	47°C
Mean speed of coolant flow inside fuel element boxes:	1.5 m/s
Mean pressure drop in core:	0.0145 MPa
Coolant water quality:	Deionized water

III. Materials Selection

Corrosion of the fuel element cladding and components inside the reactor and activation of the corrosion products can increase overall radioactivity levels in the primary loop equipment. Precipitation of corrosion products on the surfaces of equipment and fuel elements makes inspection and repair of the primary loop equipment difficult and can affect heat transfer on the element

surfaces. For this reason, we used deionized water as the coolant and used austenic stainless steel with good corrosion resistance properties for all of the materials in the core structure with the exception of the fuel elements, control rod guide tubes, beryllium blocks, aluminum blocks, ionization chamber guide tubes, and irradiation tubes.

IV. Reactor Body Structure Design

A. Overview

The entire reactor body is installed at one end in the bottom of the swimming pool. The other end serves as the fuel element storage rack. The pool is 8.5 m long, 2.2 m wide, and 8.93 m deep. The core components are installed on the lattice plate, the flange at the top of the support tube fixes the lattice plate while the flange at the lower end is fixed to the foundation at the bottom of the pool. The foundation bolts can withstand the force of a magnitude 7 earthquake. There is a 4.91 m water layer above the active parts of the core to provide sufficient protection for operations at the top of the reactor after reactor shutdowns. The primary water flows through the core from top to bottom, with about 43 percent of the flow passing through the fuel elements and 57 percent of the flow passing through the reflecting layers and other spaces. It then passes along the pool wall mother pipes and flows upward into the main pumps and heat exchangers and then flows from the top of the pool back into the swimming pool, forming an open circulation system. There is a small transport cart and a track for it below the top covering plate of the pool around the walls of the pool in the reactor position and there is a hoisting basket hanging on the cart that is pulled manually via steel cables to transfer the reactor fuel elements, beryllium blocks, and so on onto the storage racks at the side of the reactor.

B. Reactor body structure

The core rows use the matrix arrangement of the high-flux reactor core. This arrangement is the optimum program derived after comprehensive assessment of the advanced properties of physical parameters, thermotechnical hydraulics effectiveness, and structural possibilities. The entire core component rests on the lattice plate, the fuel elements are inserted into the center of the core, and there are 10 control rods placed among the elements. Outside of the fuel elements is the beryllium reflecting layer and next is the aluminum reflecting layer. The fuel element, beryllium blocks, and aluminum blocks are arranged in equilateral triangle rows based on a 64 mm lattice distance. The outermost layer is the barrel, which also serves as a heat shield. Between the reflecting layers there are monocrystalline silicon irradiation tubes or other irradiation devices of varying dimensions. The 5MW LPR body system and core configuration are illustrated in Figures 1 and 2, respectively.

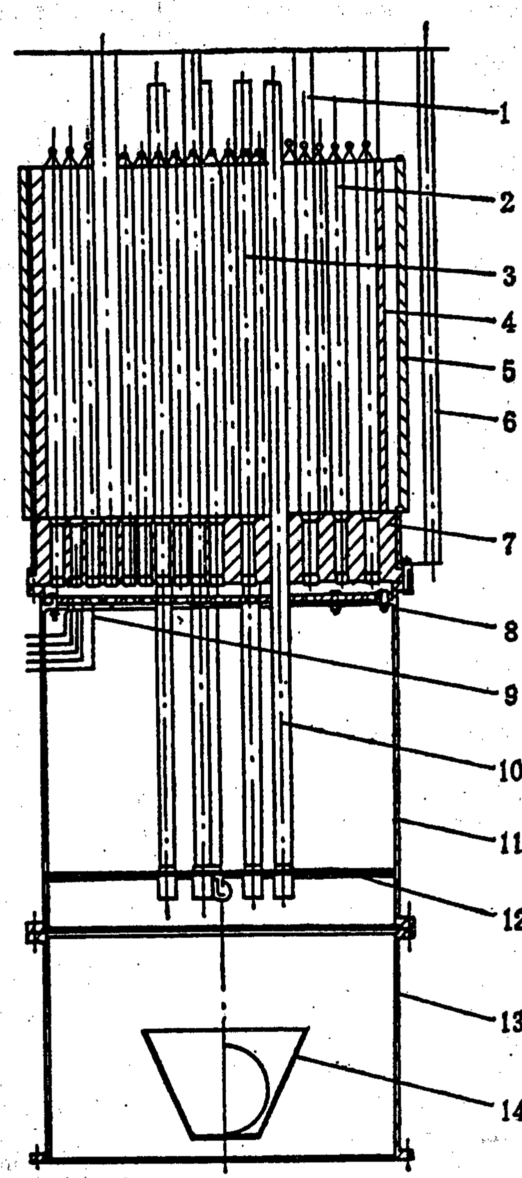


Figure 1. 5MW Reactor Body Structure

Key: 1. Irradiation devices; 2. Aluminum blocks; 3. Fuel elements; 4. Inner barrel; 5. Outer barrel; 6. Ionization chamber guide tubes; 7. Lattice plate; 8. Support plate; 9. Damage detection sampling tubes; 10. Control rod guide tubes; 11. Upper support tube; 12. Guide tube brackets; 13. Lower support tube; 14. Funnel collector

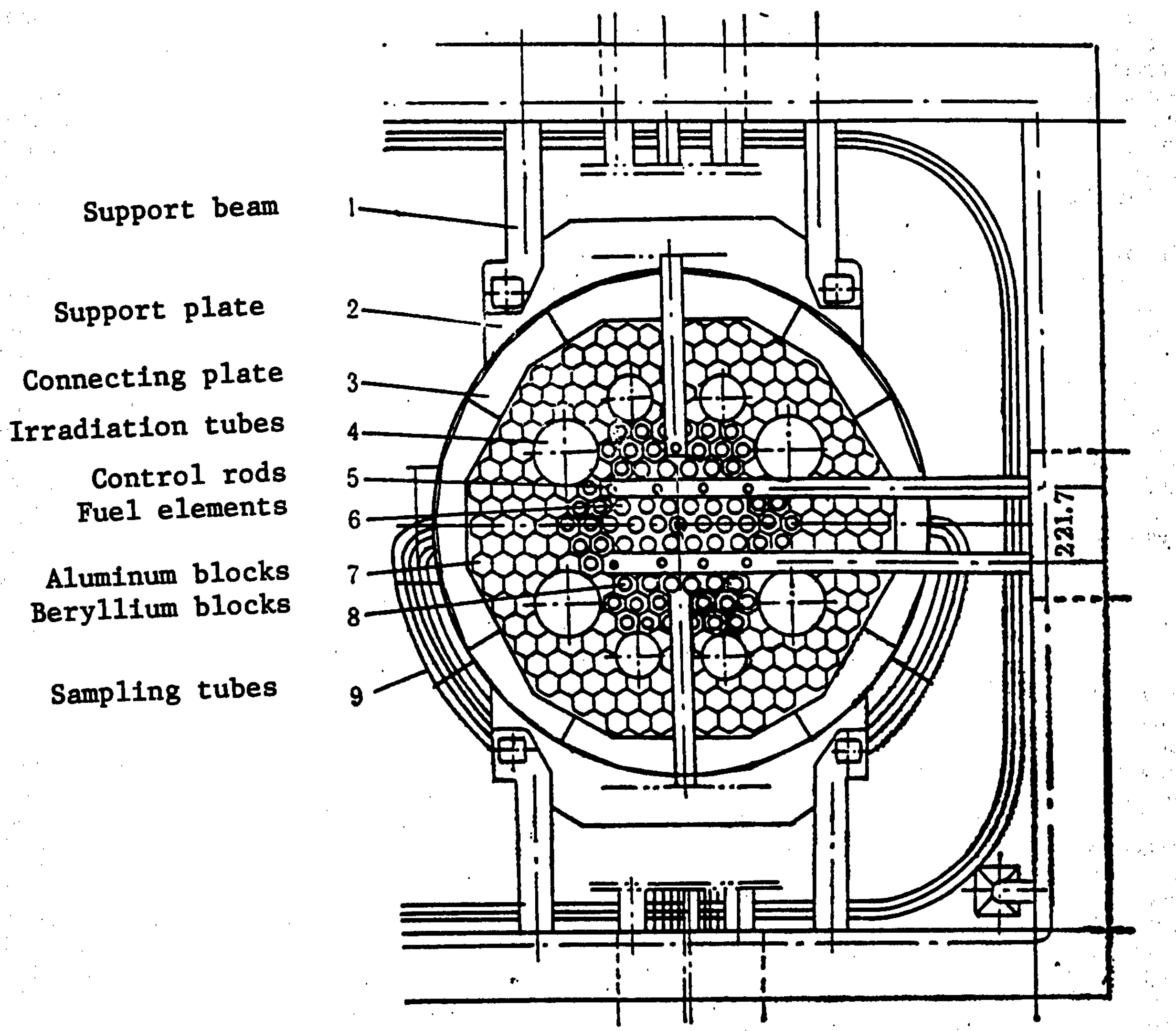


Figure 2. Configuration of Core

1. Fuel elements The fuel elements are the tubular elements unloaded from the high-flux reactor with a mean burnup of ≤ 40 percent. Their outermost layer is a φ63 x 1 mm LT24 aluminum alloy tube with a set of six inner sleeves of different diameters, and they have three ribbed uranium-aluminum alloy core fuel tubes and are clad in 305 pure aluminum. The active portion is 1 m long. The core is loaded with 32 boxes and there is a 1 mm cooling water gap between each pair of adjacent elements.

2. Control rods There are a total of 10 control rods: two automatic rods, two safety rods, and six manual rods. The rods are lifted by the pull of a drive structure and steel cables located at the top of the reactor and they can move smoothly inside the guide tubes. The aluminum control rod guide tubes penetrate the core and lattice plate and are affixed to the control rod guide tube brackets at the bottom end of the upper support tube.

3. Beryllium blocks, aluminum blocks, and stainless steel blocks These three types are rods with a hexagonal horizontal cross section and an opposite side length of 62.5 mm that form the core reflecting layers and heat shield layer. There is a water gap of 1.5 mm between each pair of adjacent rods. The rods are located around the fuel elements and there are a total of 185 blocks.

4. Irradiation devices The center of the core K_{11} is a Mo-Te [as published] irradiation duct and there are four φ120 and four φ180 monocrystalline silicon irradiation devices between the reflecting layers, including the irradiation chamber, outer sleeve tubes, and irradiation chamber rotation mechanism. The aluminum outer sleeves are the guide tubes for the irradiation chamber.

Their top ends are inserted into the core and their bottom ends are fixed to the reactor top cap plate. The irradiation chambers containing monocrystalline silicon pass through a "multisection whip"-type aluminum tube and extend into the top of the reactor and are linked to the rotation mechanism. A microcomputer and gear mechanism are used to rotate the irradiation chamber at a steady 8 rpm to ensure the homogeneity of monocrystalline silicon irradiation.

5. Lattice plate The lattice plate is one of the main components of the reactor. It was processed mechanically from an integral stainless steel casting and is 1,400 mm in diameter and 235 mm thick. It underwent two heat treatments to eliminate internal stress. The upper surface has fuel element holes, beryllium block holes, aluminum block holes, control rod guide tube holes, irradiation device holes, and so on in triangular rows, with a lattice distance of 64 mm. The lattice plate fixes and supports the entire core component. The lattice plate receives high-flux neutron and γ irradiation, and it bears the weight of the core, hydraulic shock, and the core pressure-drop load.

6. Inner and outer barrels (heat shields) To ensure the strength of the concrete organic shield and no loss of water, its working temperature is restricted to less than 100°C. To ensure that the concrete is not damaged by thermal stress, the highest temperature should be within 30°C of the surface temperature. For this reason, a 50 mm thick stainless steel inner barrel was added outside the reactor reflecting layers to reduce γ heat generation in the concrete. The inner barrel holds the core components together and also serves as a heat shield. Taking into consideration the problems with current materials and processing techniques, the barrel is divided into two layers. The inner barrel was formed in a polygonous shape and divided into 12 blocks for separate processing, and then angle iron and screws was used to connect them together. Pins were then used to affix the inner barrel to the lattice plate to facilitate hoisting and adjustment. The interior sides of the inner barrel have an irregular shape so that they conform to the shape of the contours of the aluminum blocks in the reflecting layer to prevent horizontal leakage of the cooling water. The outer barrel is a cylindrical tube with an inner diameter of 1,400 mm and a wall thickness of 30 mm, and it is affixed to the outer edge of the lattice plate with screws. The upper ends of the inner and outer barrels are connected together into one unit by several connecting plates, which also increases the rigidity of the inner barrel. The connecting plate at the upper end also increases the fluid resistance between the inner and outer barrel and prevents unnecessary leaks of the cooling water. There is a 25 mm gap between the inner and outer barrels for cooling the barrels themselves. To prevent the formation of "dead water", a notch of appropriate dimensions was opened up in the bottom of the inner barrel. Annealing to eliminate stress was done on the outer barrel to prevent seizing of the core components caused by deformation from welding stress.

7. Support tube The support tube is 1,200 mm in diameter and 1,800 mm tall. It is divided into an upper support tube and lower support tube, and the two are fixed with screws. The lattice plate is fixed directly to the upper support tube, so the support tube bears the entire weight of the core. Flange rings were welded to the inner wall of the tubes near the lattice plate end for installation of the damage detection tubes support plate. Because the welds are very near the end surface, precision processing of the flange surface was necessary after welding to prevent welding deformation from affecting the levelness of the lattice plate. The control rod guide tube support structure is at the lower part of the upper support tube. There are two φ150 damage detection pipe outlets and two φ58 auxiliary cooling outlet water holes opened in the tube walls. There is a φ325 outlet water mother pipe at the bottom of the lower support tube. The outlet water mother pipe extends into the tube and there is a funnel collector at the end of the pipe. This evenly distributes the cooling water passing through the core and does not cause "delay water tank" functions, which can reduce the ^{16}N γ dose in the primary loop. The center line of the outlet water mother pipe is elevated 250 mm above the bottom of the pool. The foundation screws connect the lower support tube and foundation into one unit and ensure that they are not damaged by the effects of heat stress or earthquake forces in the loop piping.

8. Damage detection tubes and their support structure To make a rapid determination of the location of damaged fuel elements, φ stainless steel sampling tubes were installed at the outlets on the bottom end of the 32 boxes of elements. To fix the 32 sampling tubes and four thermocouples, a 25 mm thick support plate was placed 35 mm below the lattice plate and there are holes in the plate that correspond to the lattice plate. The holes vary in size according to the flow rate. The ends of the sampling tubes and thermocouples are bent upward and extend about 10 mm into the fuel elements. These small tubes are divided into a certain number of groups and they are bent around each of the water flow holes on the support plate. Next, the tubes in each group are arranged in vertical rows and fixed to this plate with cards. All of the tubes are divided into two bundles and extend out of φ150 holes symmetrically placed on either side of the upper support tube. To reduce resistance losses inside the tubes, the φ tubes change to φ12 after leaving the support tube and are fixed in rows along the pool walls and then extend into the damage detection room.

9. Reactor nuclear measurement devices There are 12 ion chambers on the outside of the core barrels that are installed, respectively, in the 12 aluminum guide tubes. The lower ends of the guide tubes are inserted into the positioning holes in the guide tube support plate. The support plate is in two pieces that are affixed symmetrically to the outer edges of the lattice plate. The upper ends of the guide tubes are affixed to eight floating arm support beams and the support beams are welded to the top of the pool. Because of the rather large number of

positioning holes in the guide tube support plate, the guide tube positions can be adjusted as appropriate. The ion chambers are in horizontal positions inside the guide tubes and are regulated with steel cables.

Cooling System Design for 5MW Low Power Reactor

926B0125D Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 pp 23-27, 39

[Article by Zhao Yiliang [6392 0181 5328], Chen Hongzhang [7115 1347 3864], Xu Chengde [1776 2110 1795], Wang Jun [3769 0971], Zhang Guanghai [1728 0342 3189], Yu Yaping [4416 0068 1627], and Li Zhicheng [2621 1807 2052] of the China Nuclear Power Research and Design Academy, Chengdu[*]: "5MW Low Power Reactor Cooling System Design"; manuscript received 12 February 1992]

[Text] Abstract: The cooling system for the 5MW low power reactor is composed of two loops. The primary loop has closed forced circulation and the secondary loop has open circulation. The total flow rate of the water in the primary loop is 620 t/h and the total flow rate of the water in the secondary loop is 1,000 t/h. The operating power of the heat exchangers is 5MW. In the heat exchangers, the pressure of the water in the secondary loop is greater than the pressure of the water in the primary loop. Safety analysis indicates that this structure can safely and effectively remove heat from the reactor under any working conditions.

Key terms: 5MW low power reactor, cooling system, design.

I. Introduction

The 5MW LPR was rebuilt on the basis of the former 0.1MW LPR. The original 0.1MW LPR was a matching facility for the high-flux engineering test reactor (HFETR) and is a swimming pool-type light-water reactor. The design rated power is 100 kW and cooling of the reactor core relies on natural convection. The heat of the swimming pool is removed by a small cooling system (having only a single heat exchanger with a heat transfer surface of 3 m^2). After rebuilding, however, the power of the low-power reactor is 5MW, so the cooling system of the reactor had to be redesigned to ensure that the heat is effectively removed from the reactor so that the reactor operates safely and reliably.

II. Design Principles and Basic Parameters

The design principles of the 5MW LPR are:

1. The cooling system should be capable of safe and reliable operation. This means that under certain accident working conditions, it can also have the corresponding measures to ensure the safety of the reactor.

2. Every effort should be made to use the original facilities and to make full use of the HFETR design experience, experimental data, and operating experience.

3. Every effort should be made to adopt standard equipment.

4. Because of the requirement of continuous operation (about 60 days) of the reactor, attention should be given to the convenience of cooling system operation, inspection, and repair.

5. The system configuration should try to separate contaminated areas from clean areas.

6. There must be assurances that the pressure of the primary water in the heat exchangers is less than the pressure of the secondary water to prevent radioactive contamination of the secondary water.

The cooling system design parameters were decided upon on the basis of reactor thermotechnical computations (Table 1).

Table 1. Cooling System Design Parameters

Item	Value
Primary water total flow rate, t/h	620
Secondary water total flow rate, t/h	1,000
Heat exchanger operating power, MW	5
System design pressure, MPa	0.6
System design temperature, °C	≤ 60

III. Description of Reactor Cooling System

The 5MW LPR cooling system has a two-loop arrangement. The primary cooling water removes fission heat from the reactor and transfers the heat via the heat exchangers to the secondary cooling water, which then releases it via the secondary water into river water. The primary water loop passes through the swimming pool to form closed circulation, while the secondary water loop is an open system. The primary water uses high purity deionized water and the water quality indices are identical to the HFETR water quality indices. The secondary water uses filtered river water. Stainless steel is used for all of the equipment components and piping over which the primary water flows.

The 5MW LPR cooling system includes a primary water cooling system, primary water purification system, primary water replenishment system, auxiliary cooling system, emergency water replenishment system, special water drainage system, and the secondary water system (Figure 1).

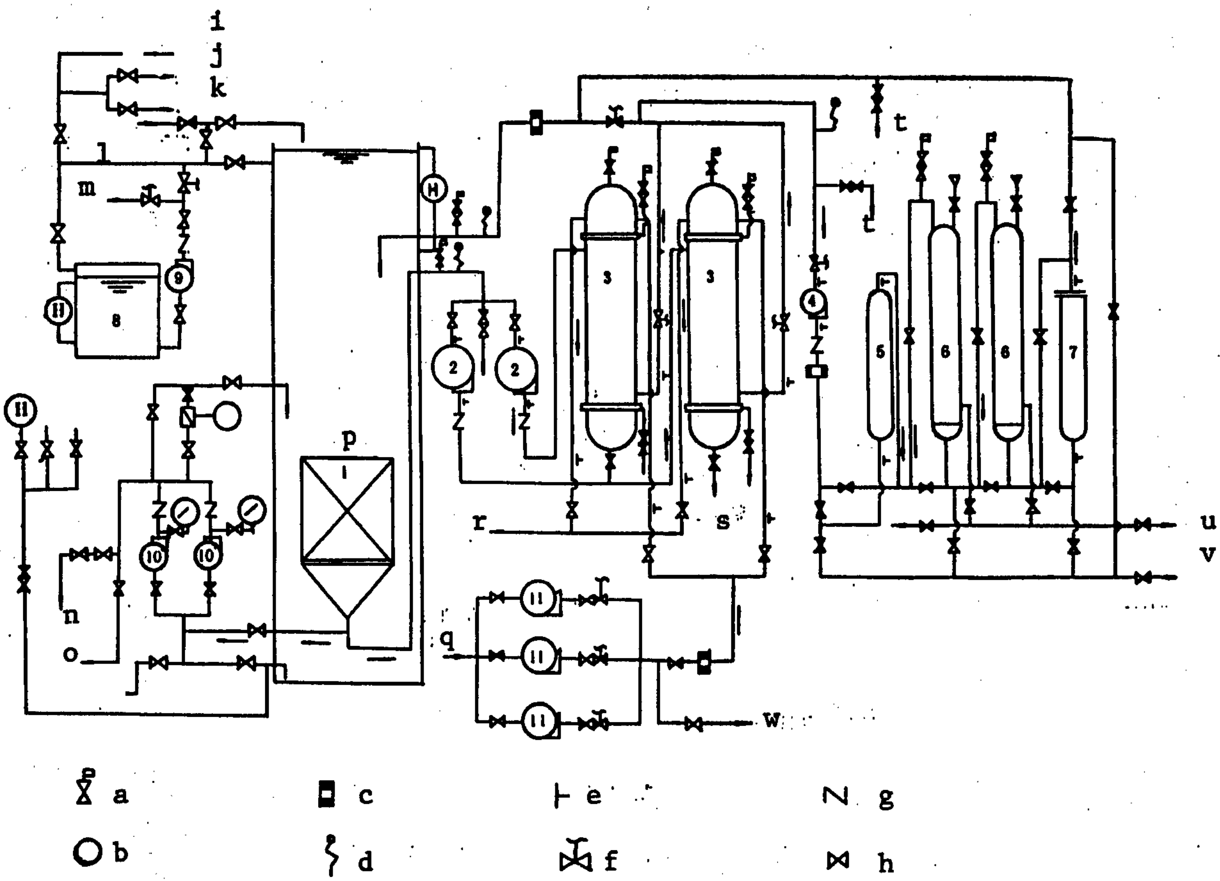


Figure 1. 5MW LPR Cooling System Flow Chart

Key: 1. Reactor; 2. Primary water pumps; 3. Heat exchangers; 4. Purification water pumps; 5. Front mechanical filter; 6. Mixing beds; 7. Rear mechanical filter; 8. Replenishment water tank; 9. Water replenishment pumps; 10. Auxiliary cooling pumps; 11. Secondary water pumps; a. Air exhaust valves; b. Water level meter; c. Flow rate duct plates; d. Temperature measurement points; e. Pressure measurement points; f. Electrical valves; g. Check valves; h. Manual valves; i. From HFETR flushing feedwater system; j. To monocrystalline silicon storage pool; k. To purification system for water used in resin transport; l. From fire control water system; m. To zero-power reactor; n. To special drainage; o. Ground leaks; p. reactor; q. From sampling water tank; r. Drained into river; s. To special drainage; t. Sampling [two locations]; u. To waste resin transport pipes; v. To special drainage; w. To HFETR small loop

A. Primary Water Cooling System

The function of this system is to carry out the fission heat from the reactor in a continuous and reliable manner under any type of working conditions (including normal reactor operation and reactor accident shutdowns).

The primary water cooling system is composed of the primary water pumps, heat exchangers, and their associated valves and piping.

The primary water flows through the core from top to bottom and carries out the heat that is in it, passing through the φ325 x 6 mm reactor outlet mother pipe across the top of the pool and into the swimming pool to the primary loop room where it splits into two parallel subloops. Each subloop has one primary water pump and one heat exchanger. In the heat exchangers, the primary water transfers the heat to the secondary water and then returns through the φ325 x 6 mm reactor inlet mother pipe and back into to the swimming pool.

Under reactor rated working conditions, the two subloops operate simultaneously and the flow rate passing through the reactor can reach 690 t/h.

There are φ25 x 2.5 mm air discharge tubes connected to the reactor inlet and outlet mother pipes at the top of the reactor pool that serve as air discharge points when the system is filled with water. They drain the loop when the primary water cooling system or the piping require inspection or repairs, or when there is a loss of a large

amount of water in the primary system, they can be used to destroy the mother pipe siphon.

The primary water pumps employ FB200-34 stainless steel corrosion-resistant pumps.

There are a total of two heat exchangers. Under rated working conditions, the two units are placed into operation simultaneously. The design of the heat exchangers adopts a mature fixed tubesheet tube matrix-type structure with a converse strike and vertical emplacement. To facilitate the washing of sediments and dirt, there is a single flow course in the primary water pipes and a dual flow course in the secondary water pipes.

B. Primary water purification system

This system is composed of one purification water pump, one front mechanical filter, two mixed ion exchange beds, one rear mechanical filter, and the associated piping, valves, and instruments.

The purification system extracts 1 percent of the primary water flow rate (6.2 t/h) from the primary water system's heat exchanger outlet mother pipe and purifies it, after which it flows through the outlet pipe into the reactor inlet mother pipe.

C. Primary water replenishment system

This system uses the water replenishment system and water replenishment tank (about 15 mm^3) from the original low power reactor. The deionized water is provided by the HFETR production water system.

Under normal conditions, the filling water and flushing and replacement water for the reactor water pool of the reactor, the primary water system, the purification system, and the monocrystalline silicon storage water pool are provided directly by the HFETR production water system. Moreover, when the reactor is operating normally the normal replenishment water in the reactor water pool is replenished from the replenishment water tank via the replenishment water pumps.

D. Auxiliary cooling system

This system is composed of the auxiliary cooling pumps and their associated valves and piping. The auxiliary cooling pumps receive their power from a reliable power source. If a full-plant power outage occurs during reactor operation, the auxiliary cooling pumps start up automatically and the primary water is extracted from the core and discharged back into the reactor water pool to ensure the removal of the residual heat generated in the core after reactor shutdown.

E. Emergency water replenishment system

When there is an unexpected loss of a large amount of water from the reactor water pool due to an accident in the reactor and the core is in danger of being exposed, emergency water injection for fire control is used to ensure safety. This system brings in the fire control water directly from fireplugs in the building at the top of the reactor pool.

F. Special water drainage system

The sources of waste water in the special water drainage system are: water drained from the reactor water pool of the reactor, water drained from the monocrystalline silicon water pool, water drained from the primary water system and purification system along with its exhaust gas and water drained for sampling, and ground leaks in the technical workshops.

Water drained from the reactor water pool of the reactor, water drained from the primary water system, water drained from the purification system, water drained for sampling, and water drained from the monocrystalline silicon storage water pool is discharged as moderately radioactive liquid waste (radioactive intensity 3.7 x 10^7 to 3.7 x 10^5 Bq/L). Water and exhaust gas that leak from the pump axial seals and ground leaks in the technical workshops are discharged as low-level radioactive liquid waste (radioactive intensity less than 3.7 x 10^4 Bq/L).

The two types of waste liquid are carried out via separate pipes connected to the original moderately radioactive and low-level special drainage mother pipes and flow by gravity into the waste water workshop.

G. Secondary water cooling system

The secondary water cooling system uses the original small loop secondary water pumps (three pumps, each with a flow rate of 360 to 612 t/h and lift of 56 to 71 m) in the HFETR three-pump building. They pass from the three-pump building to the HFETR main plant building water transmission pipes, a φ529 x 10 mm water transmission pipe near the HFETR main plant building sends the water to the low power reactor and, after it passes through the heat exchangers, out of the plant building, where it is drained into a nearby stream, forming an open loop.

At rated working conditions, two pumps are placed into operation and one pump is held in reserve. They have a design flow rate of 1,000 t/h. Ahead of and behind each heat exchanger, there is a Dg300 manual valve for isolation purposes. The outlet valves can also be used to make appropriate readjustments in the secondary water pressure in the heat exchangers.

H. Operating parameter monitoring facilities

Monitoring of operating parameters is an extremely important link in guaranteeing safe and reliable operation of the reactor cooling system. Monitoring of system operating parameters is divided into three parts:

1. Thermotechnical hydraulics monitoring. This mainly involves monitoring flow rates, pressures, and temperatures, and there are a total of 44 measurement points.

They can provide nearby or remote transmission of indicators and provide warning signals or accident signals.

2. Primary water and secondary water radioactivity monitoring (to prevent damage to the heat exchangers). This is done, respectively, by the damage detection system and contaminated water monitoring system.

3. Electrical indicators and protection.

IV. System Design Characteristics

1. The primary water cooling system uses a design that combines single element control with mother pipe control. Its advantages are: compared to total mother pipe control, it permits the reduction of four Dg200 valves and the associated piping, which saves costs and makes for a compact configuration. While it is slightly less flexible that total mother pipe control, it can ensure the safety of the reactor.

2. The primary water purification system uses a non-renewable flow process with one mixing bed operating and another mixing bed held in reserve. When regulation of the pH value is required, the reserve mixing bed can be converted to an anode bed for use. The system has a flexible design operation mode that can carry out purification during operation and do purification during periods of reactor shutdown, and the purification water can operate according to the sequence of purification pump → front mechanical filter → mixing bed → rear mechanical filter. It can also bypass the operation of any of its equipment. With the exception of the use of electrically operated valves for the inlet valves of the purification pumps, all of the manual valves in the system can be manually operated using a remote operating mechanism that penetrates the wall and runs to the purification operations workshop outside of the shielding wall. This can avoid personnel being subjected to unnecessary danger of radioactivity.

3. The primary water replenishment system and secondary water cooling system make full use of the original facilities for revision, which conserved investments.

V. System Safety Analysis

1. Under rated working conditions, the system reactor inlet flow rate is provided by two primary water pumps. When a breakdown occurs in one of the pumps, the other pump can operate independently and the reactor can operate at reduced power, during which time the system flow rate is 420 t/h.

2. The two primary water pumps are supplied with electricity from two different external power sources. When there is a breakdown in one of the external power source circuits, this arrangement can maintain continuous operation of one of the primary water pumps.

3. Each primary water pump is fitted with an inertial flywheel that lengthens the inertial rotation time of the pumps. When there is a full-plant power outage and reactor shutdown, the inertial rotation of the primary water pumps can be relied upon to maintain the core flow rate. At the same time, the auxiliary cooling system starts up and is switched in to remove the residual heat generated by the reactor and ensure that the elements are not burned up.

4. When there is a loss of a small amount of water in the system, it is replenished by the primary water replenishment system. When there is a loss of a large amount of water in the system and the liquid level in the reactor pool drops to the specified value of 500 mm, there is an accident reactor shutdown. When there is a large rupture accident in the piping, the exhaust valves in the reactor outlet mother pipe can be opened to destroy the siphon and enable the water level in the reactor pool to be maintained at a specified height.

5. The reactor inlet and outlet mother pipes are located at relatively high positions, so if there is a rupture in the mother pipes the core will not be exposed above the water surface.

6. When there is a serious loss of water that may expose the active region of the reactor, the fire control water can be injected into the reactor to ensure that there is no loss of water in the active region.

7. The damage detection system can randomly indicate the primary water total γ and fission product radioactivity.

8. When the reactor is operating, the primary water pressure inside the heat exchangers is lower than the secondary water pressure, which prevents the primary water from contaminating the secondary water.

9. Static computations for the primary water piping system indicate that the maximum stress is 78.84 MPa, which is less than the permissible stress on the piping system materials.

10. The opening and closing times of all the valves are greater than 5 seconds, so no water hammering can occur.

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Control and Protection System for 5MW Low Power Reactor

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[Article by Sun Maoyu [1327 5399 3768], Zhao Zengqiao [6392 1073 0294], Liang Yongzhi [2733 3057 1807], Fan Maixiu [2868 7796 4423], Li Guangjian [2621 0342 1696], Ma Enguang [7456 1869 1639], and Jia Guochun [6328 0948 4783] of the China Nuclear Power Research and Design Academy, Chengdu: "5MW Low Power Reactor Control and Protection System"; initial manuscript received 28 January 1992, revised manuscript received 20 February 1992]

[Text] Abstract: The 5MW low power reactor (5MW LPR) control and protection system includes: the reactor nuclear measurement system, reactor protection system, reactor control system, main control room, and so on. This reactor uses an international or domestic standard structural arrangement. Selection of components and circuits took into consideration reliability and advanced properties and the main system approximates standardization. With a precondition of satisfying safety, economy and useability were also considered. The main control room design took into consideration human engineering requirements and guarantees reactor operation environment and other requirements.

Key terms: 5MW low power reactor, nuclear measurement, control system, protection system.

I. Introduction

The 5MW LPR is a rebuilt project. Its rated power after rebuilding was increased from the original 100 kW to 5MW, and its entire control and protection system (with the exception of drive devices) had to be redesigned according to the relevant state standards. However, the control and protection system designed and processed in the 1960's lagged far behind requirements in state safety regulations. For example, the control and protection system logic circuit composed of germanium triodes could no longer satisfy the requirements for safe operation of this swimming pool-type light water reactor, so we redesigned and processed the control and protection system for the 5MW low power reactor. The operational state of the control and protection system from the time of first criticality on 3 February 1991 to successful full-power operation for 72 hours on 2 August 1991 was excellent and the design of its nuclear measurement instruments, protection system, automatic rod system, manual rod system, and so on successfully satisfied reactor operation requirements.

II. Nuclear Measurement Instruments

The nuclear measurement instruments provide the source region neutron count, nuclear power indices from startup to full-load power (counting rates, logarithmic power, linear power analog amount and digital amount), reactor power variation cycles from the source region to the power region, and signals on nuclear power and cycles that exceed specified values.

The configuration of nuclear power instruments is illustrated in Figure 1. The measurement range of the linear power measurement amplifier is 10^{-11} to 10^{-4} A. The measurement range of the broad measurement range neutron cycle monitoring amplifier is a neutron count of 1 to 10^{10} cps. The reactor cycle is 300 seconds—∞—10 seconds. The monitoring range of the power protection amplifier is 10^{-9} to 10^{-4} A. The specified value range of the power specified value regulation amplifier is 0.01 percent to 100 percent of full power.

Height regulation in the fission ionization chamber is accomplished by an electric drive with a travel of more than 700 mm. The height of the other ionization chambers can be regulated manually.

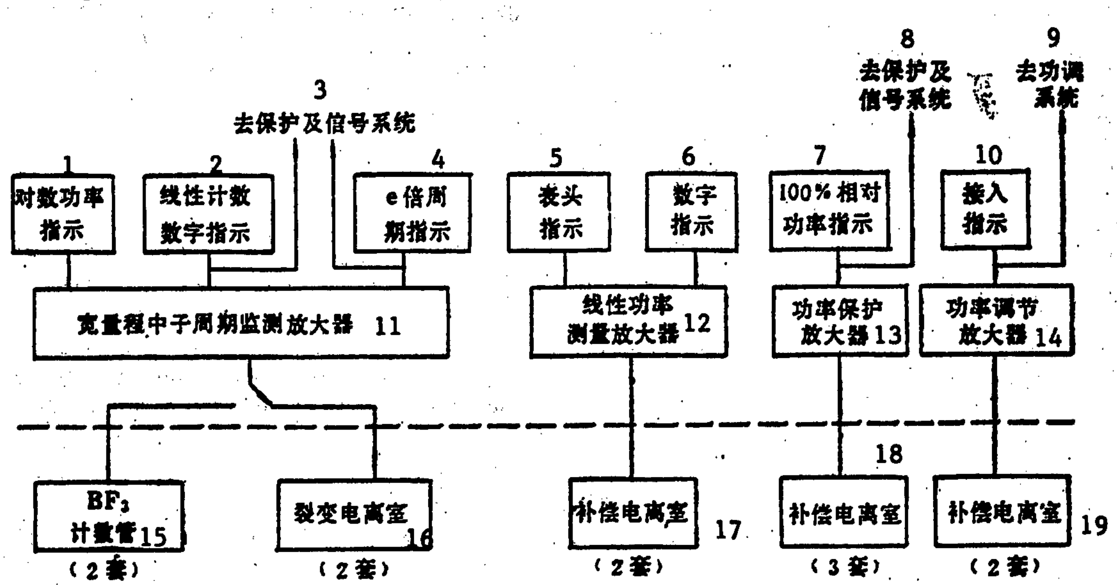


Figure 1. Nuclear Measurement Instrument Deployment Structure

Key: 1. Logarithmic power indicator; 2. Linear count digital display; 3. To protection and signal system; 4. e multiplication cycle indicator; 5. Meter indicator; 6. Digital indicator; 7. 100 percent relative power indicator; 8. To protection and signal system; 9. To power regulation system; 10. Connection indicator; 11. Broad measurement range neutron cycle monitoring amplifier; 12. Linear power measurement amplifier; 13. Power protection amplifier; 14. Power regulation amplifier; 15. BF_3 counting tube (2 sets); 16. Fission ionization chamber (2 sets); 17. Compensation ionization chamber (2 sets); 18. Compensation ionization chamber (3 sets); 19. Compensation ionization chamber (2 sets)

III. Protection System

The protection system uses CMOS assemblies with a very high degree of integration and powerful interference resistance capabilities that increased the reliability and useability of the system. Its safety logic devices and other equipment use miniature structures and its machine cases, cards, and machine frames use standardized assemblies (NIM structure) to facilitate installation and repair. Given the safety requirements of medium-sized and small reactors and economic considerations, this system has two sets of safety logic devices with common signal inputs, and the two sets of logic devices employ a one-of-two combination to guarantee safety and reliability.

A. Basis of the design

The basis of the design is 1) GB4083-83, GB4860-84, and GB5203-85; 2) Low power reactor control and protection system rebuilding requirements.

B. Safety standards

Based on the requirements in GB4083-83, the system designs focuses on consideration of these areas:

1. Single breakdown standards. The design employs redundancy technology with the objective of increasing the independence between all channels and between the protection system and the other systems so that the protection functions of the protection system can be implemented normally.

2. Breakdown safety standards. Logical "0" actions are employed. When there is a loss of power to the system, the system is in an active state.

3. Diversity of protection parameters. Monitoring of different variables is employed for the same accident occurrence to overcome breakdowns from common causes.

4. As high as possible equipment quality. Highly reliable components and parts were adopted to increase the system's reliability.

C. Safety logic devices

This protection system has two sets of completely identical safety logic devices. The two sets of devices operate in parallel in a one-of-two combination. During operation, when a breakdown occurs in one of the sets, a locked switchover switch can be used for out-of-service inspection and repair of the one set, allowing repairs without shutting down the reactor.

These devices receive signals coming from the safety monitoring devices and carry out logical processing, and provide protection action excitation signals that shut down the reactor. These devices are composed mainly of protection input elements, protection operationalization elements, terminal elements, and so on.

D. Bypass devices

Of this system's 14 circuits of accident protection signals, 10 of the circuits can be bypassed. When they are bypassed, they provide visual and audio warning signals. The set values of the power protection amplifier can be changed to implement cycle protection and pump interlocking protection or to switch out cycle protection and pump interlocking protection. There are two types of bypasses for the cycle protection, an operation bypass and a repair bypass. When both types of bypasses are placed into operation, the system will respond with an operation bypass.

E. In-service inspection devices

The role of the in-service inspection devices is to inspect safe and unsafe breakdowns in the safety logic devices during operation, to inspect the condition of the terminal elements (including the terminal relay winding), determine if the power supply to the safety logic devices is normal, and so on. Each set of safety logic devices has its own independent in-service inspection devices.

During inspection, the in-service inspection devices transmit out 11 sequential pulses that are transmitted in sequence from the 11 output ends to the 11 ends being inspected in the three protection input elements for use as inspection gate signals. When a gate signal is lost, the next one is immediately generated and two gate signals cannot be generated simultaneously. The inspection signals enter the safety logic devices from the output end of the gate. Their pulse width is 0.2 milliseconds and their cycle is 0.5 seconds. They pass through the protection input element in the logic devices and enter the protection operationalization elements and terminal elements, and the in-service inspection device counter returns from the protection operationalization elements, which causes the counter to carry continuously and change the gate position. The cycle for completing inspection of each element is about 6 seconds and they can go along with the corresponding control signals in combination to provide breakdown sound and light signals.

F. Safety warning devices

There are a total of 15 accident warning circuits (one of which is a manual emergency reactor shutdown signal). The warning indicators are red light character boards and regulation and control-type horn sounds. The 42 alarm and warning circuits are also made into independent machine boxes and placed on the protection system machine cabinet. In substance, however, they are separated and their warning indicators are yellow light character boards and DC electrical bell sounds.

When an accident occurs the red lamps light up and the horns sound. When an alarm signal occurs, the yellow lamps light up and the bells ring. The bells quiet automatically after a specific time delay and when the sound and light elimination buttons are pressed the horns cease sounding, the flashers change to normal light, and the second sound and light elimination buttons are pushed.

If the breakdown is eliminated, only then are the light signals eliminated. Otherwise, the lights do not turn off.

IV. Automatic Rod System

Two of the automatic rods in this reactor's 10 control rods can be used for manual or automatic control of reactor power. During manual operation, their rod speed is 20 mm/s. During automatic operation, they can automatically maintain the reactor power at the set level. During accidents, they automatically lower the rods in conjunction with the safety rods to shut down the reactor operation.

The design indices of the automatic power regulation system are:

1. A power regulation range of 0.01 percent to 100 percent of full power;

2. When the system quality is below a 5×10^{-4} ($\Delta K/K$) step disturbance, the amount of over-regulation is ≤ 10 percent of the set power level; the number of oscillations is ≤ 3 times; the transition process time is ≤ 3 s; the static error is < 1 percent of the set power level.

The two sets of automatic rod regulation systems are interlocked and held in reserve for each other. When one of the sets loses its primary functions, the other set automatically switches to an automatic state.

To inspect whether or not the system can be placed into automatic operation, the counter-clockwise pointing action of the automatic switches for operationalization of the systems can randomly inspect whether or not the logic functions are normal.

The logical relationship of manual, automatic, and accident rod lowering and other functions is achieved by a circuit composed mainly of CMOS assemblies. The automatic regulation system is composed of ion chambers, nuclear power regulation amplifier, silicon controlled rectifier trigger device, silicon controlled rectifier power amplifier, DC electric motor, speed reducer, speed measurement bridge-type feedback correction regulation, and automatic rods.

Besides being interlocked with the protection system, the system design also has multiple interlocks and manual and automatic interlocks between the two sets of automatic rod systems as well as automatic rod over-speed protection to prevent too great a speed of reactivity input, and so on. The rod position indication relies on selsyn-type position indicators for indication of the 1,000 mm travel.

V. Manual Rod System

The manual rods are used to compensate for the reactor's reserve reactivity. This reactor has a total of six manual rods, of which two of the manual rods have electromagnet mechanisms.

The lifting of the manual rods can be achieved only after the two safety rods are completely at the top. Second, the electric motor excitation and position indicator power sources, armature current, travel terminus, and so on are conditions that restrict the action. The circuits make full use of the interlocking between raising and lowering and the condition of lowering being given preference over raising to embody breakdown safety. The Nos 1 and 2 manual rod speeds are 3 mm/s while the other manual rod speeds are 6 mm/s, so the maximum reactivity speed that is input is always less than $2 \times 10^{-4}(\Delta K/K)$/s. This type of speed cannot cause a loss of control over the reactor or a loss of protection effectiveness. When the protection system emits a braking signal or the safety rods leave the top part, however, the manual rods will drop at 2 to 4 times the lifting speed. Added to the fact that the Nos 1 and 2 manual rods also have electromagnet mechanisms, rapid rod lowering can be achieved, taking less than 1 second to drop from the top to the bottom.

The drive portions of this system also use negative feedback speed correction to make low-speed regulation more convenient and smooth. Its control rod positions also rely on selsyn-type position indicators for indication of the 1,000 mm travel.

VI. Main Control Room

The main control room (13 m long x 7.2 m wide x 4 m tall) is located at one side of the reactor building and is isolated from it by a concrete wall over 1 m thick and a 1.9 m wide corridor.

The main things placed in the main control room are the control console, power supply cabinet, nine control screen cabinets arranged in arc-shaped symmetrical rows, and so on. Its main characteristics are:

1. The equipment in the main control room has a rational and compact layout and provides operating personnel with sufficient space; the main control room as a whole has coordinated colors and even illumination without reflections and chord light. The face of the control console has a rational layout and creates a comfortable environment for shift personnel, which helps make the best use of human and machinery capabilities and conforms to the principles of human engineering.

2. The control console uses a broadside-type arrangement. The outer edge portion of the console face has a glass cover that is used for making records and for storing charts and tables. The side panels of the control console are made of aluminum material that has undergone sand blasting processing to eliminate reflections.

3. The control console is connected to key equipment in the control portion and the protection system. Indicator instruments for important reactor parameters are installed on its vertical face. The sloping faces mainly contain various types of operating switches and indicator lamps. The display instruments for the most

important reactor parameters are placed in a central position. Some of the dials have lighted indicators for alarms, accidents, and measurement ranges that can be switched in. The important switches are placed in positions that are easy to operate but not easily touched accidentally, and when necessary the covers and eye-catching colors can be activated; the switches and manual reactor shutdown buttons for the safety rods, automatic rods, and manual rods are differentiated in function and type and the safety rod switches have keys.

4. The instruments are made of an aluminum structural material that does not require the addition of other adornments and has a natural appearance.

5. The machinery cabinets and machinery enclosures have NIM structures that facilitate their replacement, utilization, and repair.

6. There are modular structures inlaid with small magnets inside different colored glass beads in the core used for simulations that can be directly absorbed into the iron screens that are convenient and flexible during changes in loading, and the simulation charts are distinct and eye-catching.

VII. Conclusion

Operation has proven that this control and protection system has a rational design that satisfies the operating requirements of the low power reactor. However, further development and perfection is required for the application of microcomputers and advanced instruments in the reactor.

All comrades in the Institute 3 Automatic Control Room including Xie Xiangshi [6200 4161 0099] and others and some of the comrades in the Institute 1 Physics Room including Wu Manrong [0702 3341 5554] and others participated in the design, processing, installation, debugging, first criticality, and other work for this reactor's control and protection system.

Fuel Element Rupture Detecting, Locating System for 5MW LPR

926B0125F Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 pp 40-43, 48

[Article by Sun Ruixiong [1327 3843 7160] and Rao Xueming [7437 1331 2494] of the China Nuclear Power Research and Design Academy, Chengdu: "5MW Low Power Reactor Element Damage Detection and Locating System"; initial manuscript received 24 February 1992, revised manuscript received 7 March 1992]

[Text] Abstract: This article describes the element damage detection and locating system's monitoring principles, the system's configuration with 33 sampling points and their division into five groups. The sampled water passes through pumps and flow meters and is transmitted to the neutron and γ detection stations, the detected signals are transmitted to the main control room monitoring instruments and pass through grouped valves for operation to monitor the location of damaged elements.

Key terms: Fuel element damage, detection and location.

I. Monitoring Principles and Flow Processes

The 5MW low power reactor (5MW LPR) uses elements unloaded from the HFETR. To reinforce monitoring of element damage and guarantee safe reactor operation, we established a fuel element damage detection and monitoring program based on HFETR operating experience. To make it economical and reliable, local manual operation was adopted for the valves in this system. The system flow processes and sampling principles are illustrated in Figures 1 and 2.

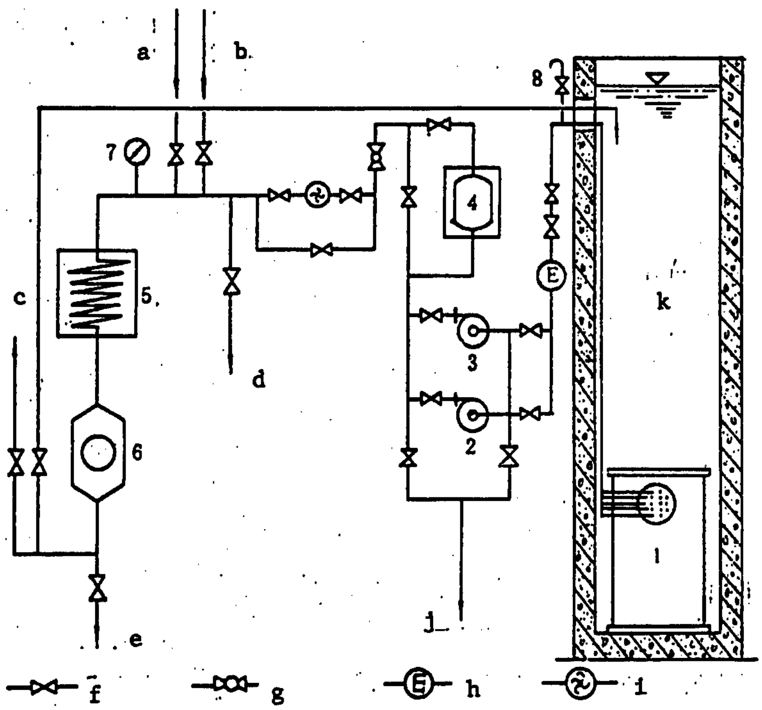


Figure 1. Diagram of Damage Detection System Flow Process Principles

Key: 1. Reactor body; 2. No 1 pump; 3. No 2 pump; 4. Delay water tank; 5. Delayed neutron detection station; 6. Total γ detection station; 7. Pressure gauge; 8. Siphon destruction pipe; a. Connection to running water; b. Connected to liquid nitrogen tank; c. To ventilation pipe; d. Sampling; e. To moderately radioactive workshop; f. Manual valves; g. Flow rate regulation valves; h. Articulation point; i. Turbine flow meter; j. To moderately radioactive workshop; k. Storage tank

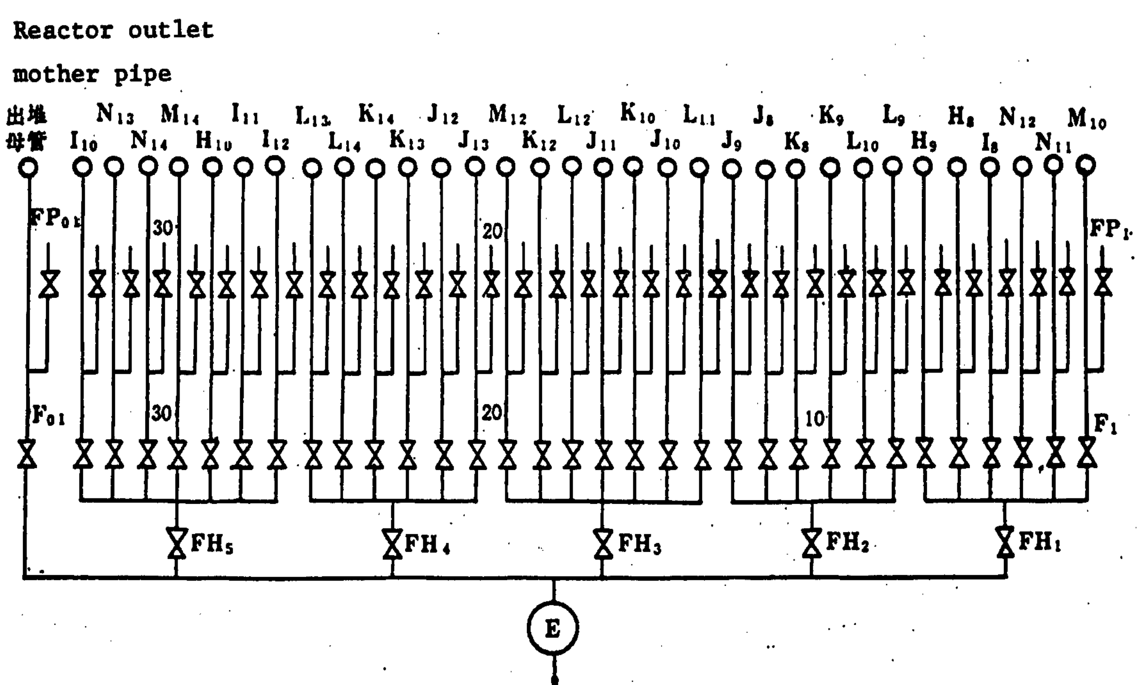


Figure 2. Diagram of Damage Detection System Sampling System

The core is loaded with 32 boxes of elements. There is a sampling pulse tube at the outlet of the element flow duct of each box that is affixed to the lattice plate, and there is also a sampling pulse tube on the reactor outlet mother pipe. These 33 sampling pulse tubes are divided into five groups and each group has these corresponding reactor elements:

Group 1: M_{10}, N_{11}, N_{12}, I_8, H_8, H_9

Group 2: L_9, L_{10}, K_9, K_8, J_8, J_9

Group 3: L_{11}, J_{10}, K_{10}, J_{11}, L_{12}, K_{12}, M_{12}

Group 4: J_{13}, J_{12}, K_{13}, K_{14}, L_{14}, L_{13}

Group 5: I_{12}, I_{11}, H_{10}, M_{14}, N_{14}, N_{13}, I_{10}

Under normal operating conditions, water samples are first monitored from the reactor outlet mother pipe, meaning that valve F_{01} is opened and FH_1-FH_5 are closed. At this point, the No 1 (or No 2) pump extracts water from the mother pipe. After passing through the delay water tank and flow meter, it enters the delayed neutron monitoring station and total γ monitoring station. When an abnormality is discovered in the counting rate transmitted by the monitoring station, the abnormality in the water sample from that group can be located through valves FH_1-FH_5, after which valves F_1 - F_{32} can be used to determine which element's sampling pipe has an abnormal water sample, thereby monitoring the location of element damage.

In Figure 2, FP_{01} and FP_1-FP_{36} are the valves used to destroy the siphon corresponding to each sampling pipe. Their functions are: 1) Drain the air in the piping; 2) Destroy the siphon. Because the reactor water pool also has a hydrostatic pressure of about 4 meters, opening these valves can destroy the siphon, which aids in repairs.

When the system and equipment are affected by contamination that affects monitoring, the valves illustrated in Figure 1 use running water or liquid nitrogen for flushing to remove the moderate radiation and the air is exhausted into the ventilation system. The sampling valves can be opened to provide water samples for radiochemical analysis.

II. System Configuration

The system configuration principles are: 1) Subject operations and maintenance personnel to the minimum possible dose; 2) Give full consideration to the adaptive capabilities of the instruments themselves to dose fields. For these reasons, the design placed equipment and pipelines with large doses in Room 115, which is isolated from Measurement Room 116 by a steel-reinforced concrete wall 30 cm thick. Isolation valve operation steel framework structures were designed for all the valves in Room 115 and isolation valve operation single-framework structures were designed for the water inlet and outlet valves in the sampling water pumps and the moderately radioactive water discharge valves to enable isolated operations in Room 116.

One end of the 32 element sampling tubes extends 10 mm downward from the lattice plate into the corresponding element flow ducts and the other end passes horizontally down along the lattice plate through the barrels on the two sides in two groups and run outward; the reactor outlet mother pipe sampling points run out of the primary loop ducts 15 cm from the bottom support tube and penetrate the pool walls of the reactor water pool at a standard height of 3.4 m and run into Room 115 in two layers of compact rows. The upper layer is fitted with the FP_{01} and FP_1-FP_{32} valves used to destroy the siphon and the lower layer is fitted with the F_{01} and F_1- F_{32} sampling valves.

The primary instruments for the FJ-1902 delayed neutron meter, the FJ-321CG5 four-channel γ alarm, and the turbine flow meter are located in Room 116. The secondary instruments are located on the main control room damage detection screen, which has two display modes: digital displays and recorders. Both the delayed neutron meter and γ alarm can transmit alarm signals to the instrument control system. Startup and shutoff operations for the water pumps can be done on the screen, as can local operations.

III. Primary Characteristics of Damage Detection Loop Design

A. System design

1. Because the sampling tubes extend out below the lattice plate, the elements are in a compact configuration at the core lattice plate, and the gaps are small (just 7 mm), the sampling tubes inside the reactor body were restricted to $\varphi 6 \times 0.5$ and there was no way to select them according to design requirements. This section has substantial resistance losses that can only be overcome by relying on selection of pump intake vacuum.

2. Because the structures were rebuilt, there was no way to open a hole for taking direct water samples at a standard height of -0.7 m. Instead, they extend from a standard height of -3.0 m to 3.4 m and then back down to the water pump room at -0.7 m, which undoubtedly increased the length of the pipes (the single length is 24 m), bending of the pipes, and the hydraulic resistance. The design resolved this problem be expanding the diameter of the sampling tubes after they exit the reactor to reduce the resistance.

3. Because this reactor is a swimming pool-type light-water reactor, the location where the switchover measurements are made in the detection pipe loop operates in the water absorption pipe section. To overcome resistance in the pipelines, the design relies on selection of pump operation performance to solve the problem.

4. Given the complexity of the switchover pipelines, to ensure that a specific measurement flow rate keeps the neutron detection station delay time within a specific

regulation range, the degree of vacuum of the pumps must be no less than 0.59 MPa. Moreover, because of the small bore of the pipes (inner diameter φ10 mm) and small flow rate (0-8 L/min), and because the liquid is radioactive primary water, the material over which the flow passes in the pumps must be stainless steel. After comparisons, the final choice was 20W-20 turbo pumps from Shenyang, which basically satisfy the design requirements for the low-power reactor damage detection system.

IV. Control Flow Rates Established For Each Type of Working Condition After Debugging

After the damage detection loop was installed and passed debugging, the system working conditions were established. To ensure that the neutron monitoring station delay time $T_{neutron} = 80$ seconds, we made computations below regarding the control flow rate q passing through the delay water tank for the reactor outlet mother pipe sampling point A' for reference in reactor production and operation. See Figure 3 for the computed parameters and symbols.

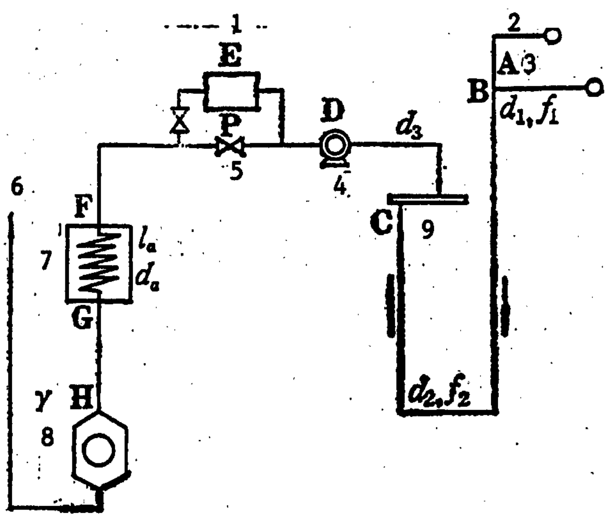


Figure 3. Illustration of Damage Detection Controlled Flow Rate Computation

Key: 1. Delay water tank V; 2. A' (Reactor outlet mother pipe); 3. A (core elements); 4. Pump; 5. Bypass valve; 6. To reactor water pool; 7. Neutron instruments; 8. γ monitoring instruments; 9. Three channels of convergent pipe

1. The time t_1 between passing through the elements and flowing to the sampling connecting tube point A': computed $t_1 = 13$ s.

2. The time t_2 for the liquid to pass through the pipeline A'BCDEF flow process: it is known that the inner diameter of the pipe $d_2 = 10^{-2}$ m, the flow passage cross section $f_2 = 7.854 \times 10^{-5}$, and the pipeline flow passing through the A'BCDEF circuit has a total length $l = 32.8$ m.

Assuming that the flow velocity in the pipe is v_2(m/s), the flow process time is:

$$t_s = l/v_2 = 32.8/v_2 \text{ (s)}$$

3. The time t_v for flowing through the delay tank: water tank volume $V = 1.8 \times 10^{-3}$ m^3, control flow rate $q = f_2v_2 = 7.854 \times 10^{-5} \times v_2$ (m^3/s), so:

$$t_v = V/q = 1.8 \times 10^{-3}/7.854 \text{ X } 10^{-5}v_2 \text{ (s)}$$

4. The time t_a to flow to the middle part of the threaded pipe at the neutron monitoring station: it is known that the inner diameter of the threaded pipe $d_a = 16$ mm and that the threaded pipe FG pipe length $l_a = 5.80$ m, so the flow velocity in the threaded pipe is:

$$v_a = (d_2d_a)^2 \times v_2 = (1016)^2 \times v_2 = 0.39v_2,$$

The time is $t_a = (1/2l_a)/v_a = 7.42/v_2$ (s)

Thus, $T_{neutron} = t_1 + t_2 + t_V + t_a = 13 + (63.14/v_2)$

When $T_{neutron} = 80$ seconds, we derive $v_2 = 0.94$ m/s, so the control flow rate:

$$q = f_2v_2 = 4.43 \text{ L/min}$$

5. The time T_r to the middle part of the water cavity at the γ detection station: it is known that the GH pipe length is 1.63 m, converting the volume of the center part of the water cavity 0.225 L into a pipe length of equivalent volume 2.86 m, thus the calculated pipe length $l_4 = 4.49$ m. The inner diameter of pipe GH $d_4 = d_2 = 10$ mm, so $v_4 = v_2 = 0.94$ m/s, and the time to flow through l_4 is $t_4 = l_4/v_4$, so we can derive

$$T_r = T_{neutron} + t_a + t_4 = 93\text{s}$$

The results of the above calculations of the control flow rate basically satisfy the measurement requirements.

V. Conclusion

Comparing this design to the HFETR damage detection system, because it involves rebuilding and is subject to restriction by objective conditions, there were rather substantial problems in technology and installation. Nevertheless, because of efforts in the area of the design and by means of debugging, several technical problems were resolved and it received approval from the National Nuclear Safety Administration which witnessed the debugging, which enabled the damage detection system to satisfy the measurement requirements in an initial way.

Fuel Element Transport Within Plant for 5MW Low Power Reactor

926B0125G Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 pp 44-47

[Article by Zhang Zicai [4545 1311 2088], Yao Shibin [1202 1102 1755], Tang Xueren [0781 1331 0088], Huang Mingtai [7806 2494 1132], and Dong Yusheng [5516 3254 3932] of the China Nuclear Power Research and Design Academy, Chengdu: "In-Plant Transport of Fuel Elements for the 5 MW Low Power Reactor"; initial manuscript received 12 February 1992, revised manuscript received 20 February 1992]

[Text] Abstract: This article describes the in-plant transport technique and its safety analysis for the dry transport technique used to move the fuel elements unloaded from the HFETR to the 5MW low power reactor (5MW LPR). It provides the results of practice in transporting the 32 boxes of fuel elements unloaded from the HFETR for the first heat loading of the 5MW LPR.

Key terms: 5MW low power reactor, unloaded elements, lead cask, forklifts, in-plant transport.

I. Introduction

The 5MW LPR uses the elements unloaded from the high-flux engineering test reactor (HFETR) with a burnup of less than 40 percent for its fuel. Because the two reactors are not in the same building, the unloaded elements had to transported via a roadway in the plant area. Usually, lead casks are used as containers for transporting unloaded elements, which is done using a wet transport method. Lead casks used for highway and railway transport of elements unloaded from the HFETR are now being developed. To meet the needs of the low power reactor rebuilding project and deal with the characteristics of the short travel involved in in-plant transport and the full safety guarantee conditions, an underwater loading and unloading and dry transport technique was adopted, and we designed and manufactured a structurally simple lead transport cask that can transport three boxes of elements at a time. We also used a forklift with a low center of gravity and convenient cargo loading and unloading as the transport vehicle. This transport technique passed a safety evaluation by the National Nuclear Safety Administration in 1990 and successfully moved the 32 boxes of elements unloaded from the HFETR for the first heat loading of the 5MW LPR from the high-flux reactor plant building safely to the 5MW LPR plant building.

II. Transport Technique

The in-plant transport technique for the 5MW LPR fuel elements adopted an underwater loading and unloading and dry transport technique. The elements were loaded into and unloaded from the lead cask underwater. During transport, the interior cavity of the lead cask was not filled with water and the elements were in a dehydrated state. The transport technique process is:

1. The bolts holding the top cover of the unloaded lead cask are removed and a hoist in the high-flux reactor building is used to lift the lead cask into the element storage pool.

2. The top of the lead cask is opened underwater and special tools are used to load the elements into the lead cask. After the top cap has been put in place, the lead cask is hoisted out of the water and the water that accumulated inside the cask is drained out.

3. After flushing the surface of the lead cask and tightening its top cap, the lead cask is hoisted and transported into the transport bottom case.

4. The forklift hoists the bottom case along with the lead cask and, after it is tied down, transports it to the 5MW LPR plant building where it is unloaded.

5. The bolts holding the top cap are removed and a hoist in the 5MW LPR plant building lifts the lead cask into the reactor water pool.

6. The top cap is opened underwater and special tools are used to remove the elements from the lead cask. After affixing the top cap, the lead cask is hoisted out of the water and the water that accumulated inside the cask is drained out.

7. After flushing the surface of the lead cask and tightening its top cap, it is returned in the manner described above back to the high-flux reactor plant building, thus completing one transport cycle.

III. Transport Container and Vehicle

A. Lead cask

The lead cask was the container used to transport the unloaded elements. Its design capacity is three boxes of elements per cask. Based on the phased and intermittent operation working conditions of the HFETR in the past several years, the shielding computations assumed that the box power of the elements in the final operation phase would be 1,000 kW, that it would operate continuously for 25 days, and that cooling the reactor after it was shut down would take 1 year. The primary technical parameters for the lead cask are listed in Table 1. The structure is illustrated in Figure 1.

Table 1. Primary Technical Parameters for Lead Cask

External dimensions, mm		φ580 x 1920
Internal cavity dimensions, mm		φ147 x 1530
Elements loaded and transported, boxes		3
	Radial, cm	20.0
Lead layer thickness	Top, cm	19.0
	Bottom, cm	15.5
Surface dose rate, mSv/h		< 0.4
Weight, t		5

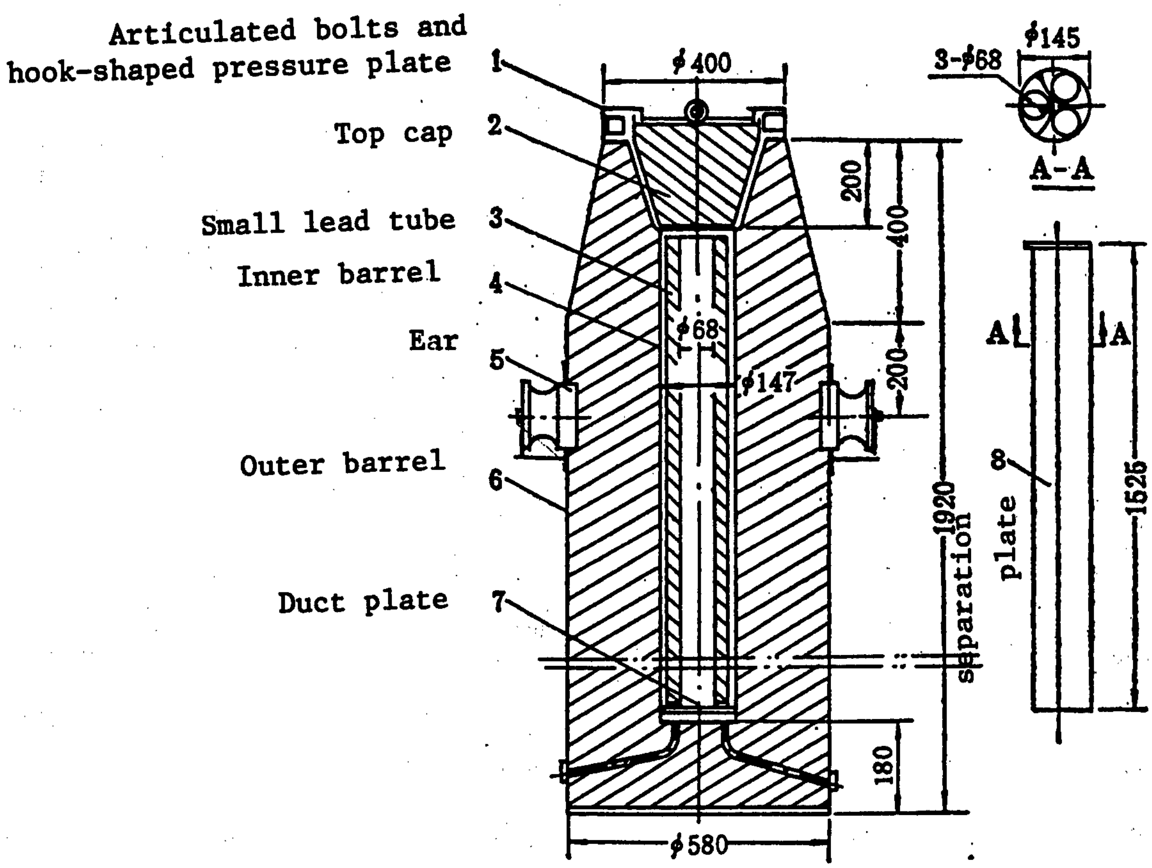


Figure 1. Lead Cask Used To Transport 5MW LPR Fuel Elements

The lead cask is composed of lead poured between inner and outer steel drums. The top is affixed with four M16 articulated bolts that hold the top cap with hook-shaped pressure plates. There is a water drainage hole in the bottom. The inner cavity has positioning separation plates to separate the elements. There are three element insertion holes in the top end of the separation plates with three color indicators, red, blue, and yellow, to differentiate the positions of the elements in the lead cask. To expand the scope of utilization of the lead cask, we designed a small lead tube that can be placed into the lead cask instead of the positioning plates. Using the small lead tube increased the thickness of the lead layer for loading and transporting powerful unloaded elements or isotopes.

After it was fabricated, the lead cask underwent γ radiation leakage tests and the shielding quality attained design requirements.

B. Transport vehicle

Forklifts have advantages like a large weight of their own, low center of gravity, ease in loading and unloading cargo, and so on, and are widely used in cargo loading and

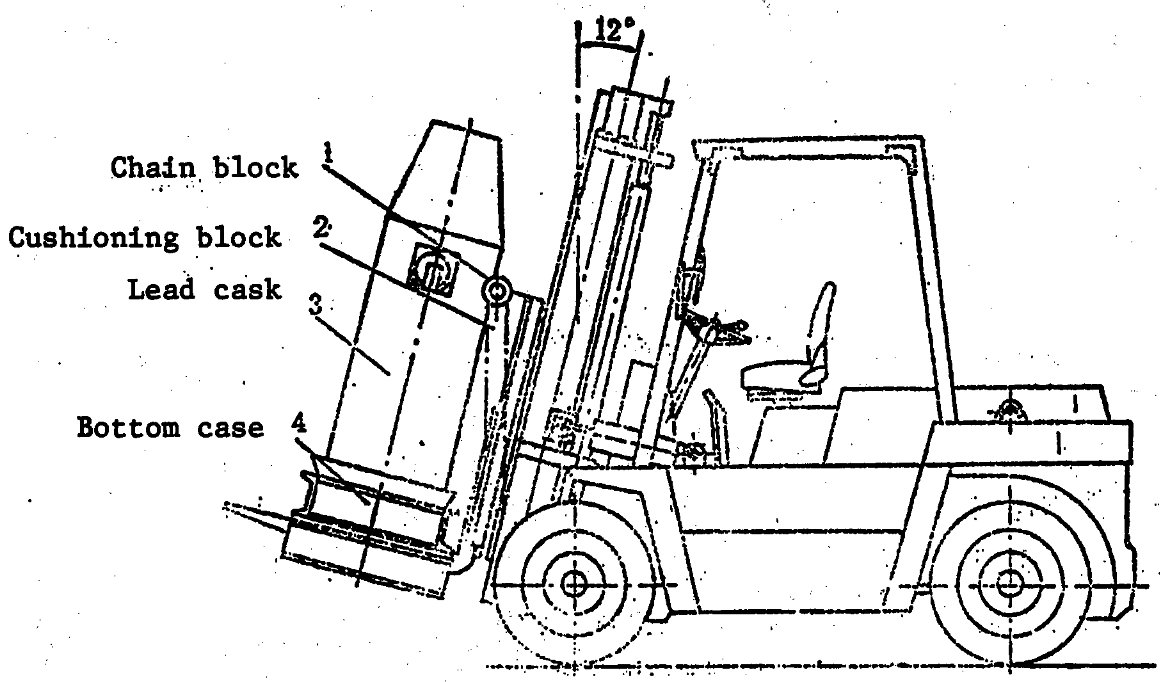


Figure 2. Forklift Transporting Lead Cask

unloading and short-distance transport in industrial and mining enterprises. A Chinese-made model CPCD-60A forklift with a loading weight of 6 tons was selected for in-plant transport of the fuel elements for the 5MW LPR. The method used to transport the lead cask by forklift is illustrated in Figure 2. The lead cask sits on a bottom case welded from channel steel. The upper part is attached to the cargo rack on the forklift by a chain block. The middle part is positioned using a V-shaped wood cushioning block. The loading operation is easy and quick.

IV. Transport Safety Analysis

A. Critical safety

The effective multiplication coefficient of the seven boxes of 40 percent burnup elements unloaded from the HFETR is $k_{eff} < 0.7$. During transport, the lead cask was loaded with and transported three boxes, so there was no critical safety problem.

B. Radiation protection analysis

1. All loading and unloading operations for the unloaded elements are done out in water pools at the two reactors. The protective water layer is more than 5 m thick, so it provides sufficient safety.

2. After the lead cask is drained of water and is surface is flushed, the residual water droplets collect in the transport bottom case, permitting contamination of the surface of the lead cask and the environment to be controlled.

3. Shielding computations indicate that the surface dose rate of the lead cask during transport does not exceed 0.4 mSv/h. The 5MW LPR has 32 boxes of elements loaded for each heat, so it required 11 transports, for a total of 44 transports for four heats during a year of operation. Because the lead cask loading and unloading operations during the transport process are easy and quick, calculating on the basis of work personnel spending 15 minutes for each transport while operating at the surface of the lead cask, the calculated annual total dose received is about 4.4 mSv, far below the permissible value stipulated by the state for radiation protection safety.

C. Temperature of the unloaded elements inside the lead cask

Computing the decay heat power for a box power of 1,000 kW and continuous operation for 25 days for the elements unloaded from the HFETR after cooling for 1 year after reactor shutdown, we obtained a residual power of 31 W per box of elements, which is about 111.6×10^3 J/h. The box heat capacity of the HFETR elements is 2.9×10^3 J/h. Assuming that the elements inside the lead cask are in a colored heat state and that all the residual heat released from the elements is used to heat the elements themselves, the temperature rise speed of the elements is 38.5°C/h. Actually, most of the energy from the residual heat generated by the elements is absorbed by the lead cask after γ rays penetrate the elements, so the temperature rise speed of the elements will not exceed 20°C/h. The dehydration time of the elements during the transport process will not exceed 2

hours, so the temperature of the elements inside the lead cask will be below 80°C and thus cannot cause element overheating problems.

D. Transport safety

The total weight of the lead cask and transport bottom case is about 5.2 t and the lifting weight of the forklift is 6 t, so there is a 20 percent overload capacity and the lifting safety margin is as much as 38 percent. Moreover, the load center distance of the forklift is 600 mm and the load center after loading and transporting the lead cask is less than 400 mm, thus ensuring the stability of the forklift while being driven. The lifting height for the lead cask during transport was restricted to no more than 0.5 m and the driving speed was < 10 km/h, which further improved transport safety.

E. Accident analysis and safety assurances

The route taken for in-plant transport of the 5MW LPR fuel elements is illustrated in Figure 3.

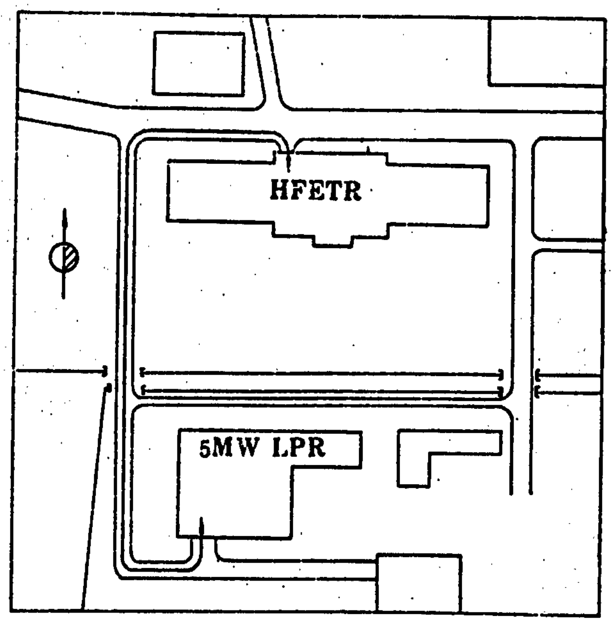


Figure 3. Diagram of Transport Route for In-Plant Transport of 5MW LPR Fuel Elements (travel about 200 m)

One can see in Figure 3 that the transport route is not a primary communication trunkline for the plant region, there are extremely few vehicles driving there, the roadway over the course (about 200 m) is level, there are no bridges or precipices, and the structures along either side contain no flammable or combustible materials. Thus, the possibility of collisions, turnovers, fires, or other accidents is extremely small. Accidents like the lead cask dropping off, being burned, and so on could occur, but because the height it would drop is relatively small and because the lead cask also has adequate strength and the top cap is affixed with bolts, the elements could not fall out of the lead cask or be crushed. For short-duration fire accidents, we rely on the enormous heat capacity of the lead cask, so they likewise would not endanger the safety of the elements.

To ensure transport safety, we also implemented a series of safety assurance measures. They included technical training for work personnel, pre-transport equipment inspections, transport experiments with the empty lead cask, communication management along the transport circuit, strengthening fire prevention safety measures, and so on.

In summary, because of the safe and reliable equipment, full safety assurance conditions, and fixed posts and technical proficiency of the work personnel involved in in-plant transport of the 5MW LPR fuel elements, transport safety could be reliably guaranteed. If an accident did occur, it could be quickly eliminated within the scope of the plant region so that the harm from an accident could be controlled to a minimum.

V. Transport Practice and Conclusions

The 32 boxes of elements unloaded from the HFETR for loading the first heat in the 5MW low power reactor were completely and safely moved from the high- flux reactor plant building to the 5MW LPR plant building in January 1991. The average burnup of this group of elements was 38.42 to 40.00 percent. They had cooled for about 1.5 years after the reactor was shut down. Three boxes of elements were transported each time in the lead cask. The measured surface dose rate of the lead cask was less than 0.25 mSv/h. It took less than 1.5 hours to complete one transport and the dehydration time of the elements in the lead cask was less than 0.5 hours. The results of operation practice at the 5MW LPR confirm that the elements were of excellent quality and exhibited no indications of damage.

Practice during in-plant transport of the elements unloaded from the HFETR showed that adopting underwater loading and unloading and dry transport was technically rational, required few investments in equipment, and was technically safe and reliable. At present, wet transport is usually used to transport unloaded elements. The dry transport described in this article is a new attempt and its practice and experience has very good reference value for rational selection of spent fuel element transport techniques.

The fabrication and inspection work for the lead cask used in this project received substantial assistance from comrades Tu Shunqing [3205 7311 3237], Li Xingheng [2621 2502 1854], Jiang Jiashun [5592 0163 7311], and others, and we would like to express our gratitude here.

Bulk Shielding Design, Safety Analysis of 5MW Low Power Reactor

926B0125H Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 pp 49-54

[Article by Fu Shouxin [0265 1343 0207], Yu Weide [0060 4850 1795], and Liu Guilian [0491 2710 5571] of the China Nuclear Power Research and Design Academy, Chengdu: "5MW Low Power Reactor Bulk Shielding Design and Safety Analysis"; initial manuscript received 28 January 1992, revised manuscript received 25 March 1992]

[Text] Abstract: This article describes radiation field and temperature field computation models, methods, and programs for the bulk shielding of the 5MW low power reactor (5MW LPR). It provides the primary computation results for the revised design. The incident neutron flux and γ flux on the inner surface of the concrete shield both satisfy stipulated design standards and the maximum temperature, maximum temperature rise, and maximum temperature gradient all conform to stipulated requirements. To confirm the reliability of the temperature field programs, the computed values and actual measured values for the temperature inside the concrete shield of the HFETR were compared and the results conformed very nicely.

Key terms: 5MW low power reactor, bulk shielding, source intensity density, maximum temperature rise, maximum temperature gradient, homogenization.

I. Introduction

The 5MW LPR is a matching project of the HFETR and its fuel is the elements unloaded from the HFETR. The core design thermal power is 5MW. It is used mainly for monocrystalline silicon doping and isotope production.

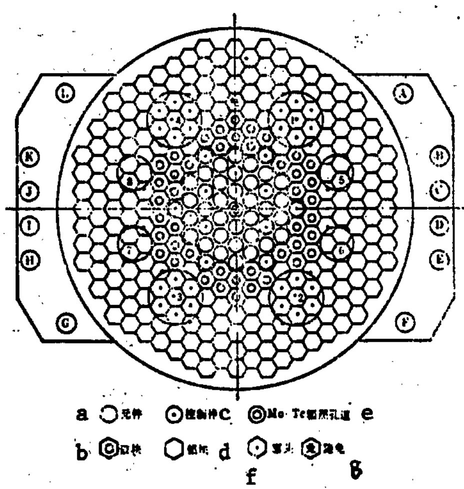
The goal in these calculations is to ensure radiation safety during production operation and the irradiation safety of work personnel, thereby completing safety assessment requirements.

The content of the computations includes radiation field computations (including monocrystalline silicon irradiation ducts and ionization chamber ducts) and temperature field computations for the inner surface of the concrete bulk shielding to enable determination of the incident fast neutron flux and γ ray flux on the inner surface of the concrete and the maximum temperature, maximum temperature rise, and maximum temperature gradient inside the concrete shielding layer. All of these values must satisfy the provisions of the "Pressurized-Water Reactor Nuclear Power Plant Radiation Shielding Design Standards"[1].

It was discovered after making calculations for the original design of the reactor body structure that the incident γ energy flux on the inner surface of the concrete on the side nearest the core exceeded standard stipulations. Following the revision, after some of the aluminum blocks in the outer margin of the aluminum reflecting layer were replaced with stainless steel blocks (meaning the addition of heavier material), the computed results satisfied the stipulated requirements.

II. Reactor Body Structure

The 5MW LPR core is composed of 32 unloaded elements with an average burnup of about 38 percent, one Mo-Tc irradiation duct, and four control rods arranged in hexagonal rows, and the hexagonal lattice element has an area of 35.47 cm^2. Around them in order are the beryllium reflecting layer (containing six control rods), the aluminum reflecting layer (containing eight monocrystalline silicon ducts), the inner barrel, the water layer, the outer barrel, the stainless steel sleeve, and the concrete shield layer. The structural configuration is illustrated in Figure 1.



1,2,3,4 - ϕ180 monocrystalline silicon ducts
5,6,7,8 - ϕ120 monocrystalline silicon ducts

Ducts: A,B,C,D,E,F,G,H,I,J,K,L - ionization chamber ducts

Figure 1. Vertical View of 5MW LPR

Key: a. Elements; b. Beryllium blocks; c. Control rods; d. Aluminum blocks; e. Mo-Tc irradiation ducts; f. Plugs; g. Running rabbit

The core is effectively converted into five regions according to the different materials used: a Mo-Tc irradiation region, first fuel element region (burnup 37.15 percent), control rod region, second fuel element region (burnup 38.72 percent), and third fuel element region (burnup 40.88 percent). Their effective outer radii in sequence are 3.20 cm, 8.89 cm, 11.14 cm, 14.65 cm, and 20.44 cm. The height of the active section of the core is 100 cm.

The homogeneous beryllium reflecting layer is made of 44 beryllium lattice elements and six control rod lattice elements. The lattice element areas are all approximately equivalent to the element lattice element area with an effective radius of 31.34 cm.

The aluminum reflecting layer is made of 185 aluminum lattice elements and 40 monocrystalline silicon lattice elements with an effective radius of 59.45 cm.

The material used for both the inner and outer barrels is stainless steel with a density of 7.86 g/cm^3 and their effective thicknesses are, respectively, 3.03 cm and 3 cm.

The material used for the steel inner sleeve of the concrete shield is also stainless steel. The minimum distance between the steel inner sleeve and outer barrel is 27.5 cm. The concrete shielding layer is 280 cm thick and is made of common concrete.

The distance between the surface of the core and the surface of the water at the top of the reactor is 541 cm and the distance between the water surface and the outer wall of the aluminum capping plate is 66 cm.

III. Design Standards

These computations are based on the Ministry of Nuclear Industry's "Pressurized-Water Reactor Nuclear Power Plant Radiation Shielding Design Standards". To restrict the effects of nuclear heat generation and ensure the stability and integrity of the shield body, the common silicate concrete shield body should satisfy the following requirements:

1. A neutron flux density on the inner surface of less than or equal to 5×10^9 $n/cm^2 \times s$;

2. A γ ray flux density on the inner surface of less than or equal to 4×10^{10} $MeV/cm^2 \times s$;

3. A maximum temperature gradient of less than 100°C/m;

4. A maximum internal temperature rise of less than 6°C;

5. A maximum internal temperature of 85°C (for shielding from neutrons) or 175°C (only used for shielding from γ rays).

IV. Parameter Computations

A. Fission source intensity density distribution

The ANISN program[2] concerning the fixed source problem requires that the fission density distribution of the element region be plotted. The fission density can be obtained from the power density distribution or neutron flux distribution provided by core physical computations. When the power density distribution is known, the source intensity density can be derived using the following formula:

$$S(r) = K \times v \times P(r)$$

In the formula, K is the fission count per second per Watt for the ^{235}U, 3.1×10^{10} $W^{-1} \times s^{-1}$; v is the average second generation neutron count released by each instance of fission, and is assumed to be 2.5; P(r) is the core power density distribution, W/cm^3.

When the core neutron flux distribution is known, the source intensity can be derived using the following formula:

$$S(r) = \sum_g \nu_g \Sigma_{f,g} \phi_g(r)$$

In the formula, $\Sigma_{f,g}$ is the macro fission cross section of the gth group, in cm^{-1}; $\varphi_g(r)$ is the neutron flux for the gth group in the fission material region, in $n/cm^2 \times s$; v_g is the number of second generation neutrons emitted from each instance of fission for gth group neutrons.

B. Nuclear density

In the data provided, homogenization processing has already been carried out for the nuclear density of each lattice element in all regions of the core.

Because the beryllium reflecting layer contains control rods and the aluminum reflecting layer contains monocrystalline silicon, homogenization processing must also be carried out. The homogenized nuclear density of the beryllium reflecting layer takes into consideration the volume ratio occupied by the six control rods (in a fully lifted rod state); the homogenized nuclear density of the aluminum reflecting layer takes into consideration the volume ratio taken up by the four φ180 and four φ120 monocrystalline silicon irradiation ducts, and the monocrystalline silicon takes up the area roughly of 40 lattice elements.

V. Computation Methods and Programs

This work used the internationally accepted discrete coordinate (Sn) method to compute the bulk shielding neutron and γ ray flux distributions. Its programs are ANISN (one dimensional)[2] and DOT-3.5 (two dimensional)[3]. The cross section library used the BUGLE-80 library[4]. It is a cross section library suitable for the 67 group $(47n + 20\gamma)P_3$ expansion used for shielding computations. P_3S_8 approximation was used in the computations.

First, we used the ANISN program to compute the group flux distribution for the shielding system (the computation models are given in Table 1). We used this to combine 13 groups of neutron and 4 groups of γ ray regions into related weighted cross sections. Then, we used this small group of weighted cross sections to do one-dimensional computations and two-dimensional DOT(r,z) computations.

Table 1. Computation Models

Sequence	Region	Radius, cm	Thickness, cm	Grid number	
				One dimensional	Two dimensional
1	Molybdenum-technetium region	3.2	3.2	2	2
2	First element region	8.89	5.69	5	5
3	Control rod region	11.14	2.25	2	2
4	Second element region	14.65	3.51	3	3
5	Third element region	20.44	5.79	6	6
6	Beryllium reflecting layer	31.34	10.9	5	15
7	Aluminum reflecting layer	59.45	28.11	15	19
8	Inner barrel	62.48	3.03	5	5
9	Water layer	67.0	4.52	3	4
10	Outer barrel	70.0	3.0	3	5
11	Water layer	97.5	27.5	14	20
20	Steel sleeve layer	97.8	0.3	1	1
13	Concrete layer	377.8	200.0	70	10

VI. Results of Radiation Field Computations and Analysis

Computations of the neutron and γ ray flux on the surface of the inner wall of the concrete shield were done using the two-dimensional DOT(R,z,) model. We used 13 groups of neutron and 4 groups of γ ray weighted cross sections and P_3S_8 approximation. The group structures of the 13 groups of neutron and 4 groups of γ rays are given in Table 2.

Table 2. Neutron and γ Ray Energy Group Structures and Fission Spectra

Energy group	Neutrons			γ rays	
	Energy upper limit, eV	Lux upper limit	Fission spectrum, MeV^{-1}	Energy Group	Energy upper limit, MeV
1	1.7333E7	-5.5000E-1	0.512593E-1	1	14.00
2	4.9659E6	7.0000E-1	0.150342	2	6.00
3	3.0119E6	1.2000	0.860718E-1	3	3.00
4	2.4660E6	1.4000	0.458223E-1	4	0.80
5	2.2313E6	1.5000	0.339513		
6	1.0026E6	2.3000	0.136923		
7	6.0810E5	2.8000	0.152048		
8	1.1109E5	4.5000	0.181216E-1		
9	3.3546E3	8.0000	0.312437E-3		
10	4.5400E2	1.0000E1	0.371931E-4		
11	1.0130E2	1.1500E1	0.101202E-4		
12	5.0435	1.4500E1	0.486407E-6		
13	4.1399E-1	1.7000E1	0.434951E-7		

The highest neutron flux and γ ray flux are located at the height of the plane of the core.

According to calculations in the original program, the total neutron flux on the surface of the inner wall of the concrete is 3.174 x 10^8 n/cm^2 x s and the total γ energy flux is 2.051 x 10^{11} Mev/cm^2 x s.

Most of the structural materials in this reactor are light materials. Very few heavy materials are used for effective shielding of γ rays. There is only 6.33 cm of stainless steel between the surface of the sides of the core and the concrete and steel inner sleeve. The other materials are 32.02 cm of water, 28.1 cm of aluminum, and 10.9 cm of beryllium. The total area mass number is 177.28 g/cm^2. Weakening 3 MeV photons then by about 10 times requires an area mass number of 60 g/cm^2. For this reason, the total area mass number described above weakens the γ rays by about 3 quantum grades. In the core, the average γ ray flux with energies in the range of 0.8 to 3 MeV is 9.64 x 10^{13} Photon/cm x s, so the γ flux at the 10^{11} quantum level on the inner surface of the concrete is believable. However, this result exceeds by 4 times the limits stipulated in Ministry of Nuclear Industry standard EJ-317. If this is the case, the maximum temperature rise from nuclear heat generation may exceed 6°C and thermal stress may be created in local areas by an excessively large temperature gradient, which could cause fracturing of the concrete or the formation of cavities and reduce its shielding performance.

Because of the relatively small amount of heavy materials used in the 5MW LPR, the main contribution to the γ incident flux on the surface of the inner wall of the concrete comes from primary γ rays generated by the core. For this reason, part of the aluminum blocks in the aluminum reflecting layer on the side nearest the inner surface of the concrete had to be replaced with stainless steel material. Computations established that the two layers of aluminum blocks in the outermost part of the aluminum reflecting layer within a flare angle range of about 40° had to be replaced with stainless steel blocks. This type of revision must satisfy these two requirements:

1. It must reduce the incident γ flux on the inner surface of the concrete to below the stipulated limit;

2. To ensure irradiation efficiency, the thermal neutron flux at the site of the monocrystalline silicon ducts must not be reduced too much by the addition of the stainless steel.

The final program decided upon satisfied these two requirements. Figures 2 and 3 show, respectively, the neutron and γ ray radial flux distribution curves. The total neutron flux on the surface of the concrete after the program revision is 1.096 x 10^8 n/cm^2 x s. The total γ ray flux is 2.854 x 10^{10} MeV/cm^2 x s. The decrease in the thermal neutron flux in the monocrystalline silicon ducts after the revision was not very great. Table 3 lists the γ incident flux on the inner surface of the concrete after the revision and the thermal neutron flux in the central part of the monocrystalline silicon and ionization chamber irradiation ducts.

The neutron and γ ray flux distribution in the concrete shield were calculated using the one-dimensional ANISN program and the internal heat source distribution inside the concrete was computed from the γ ray flux distribution.

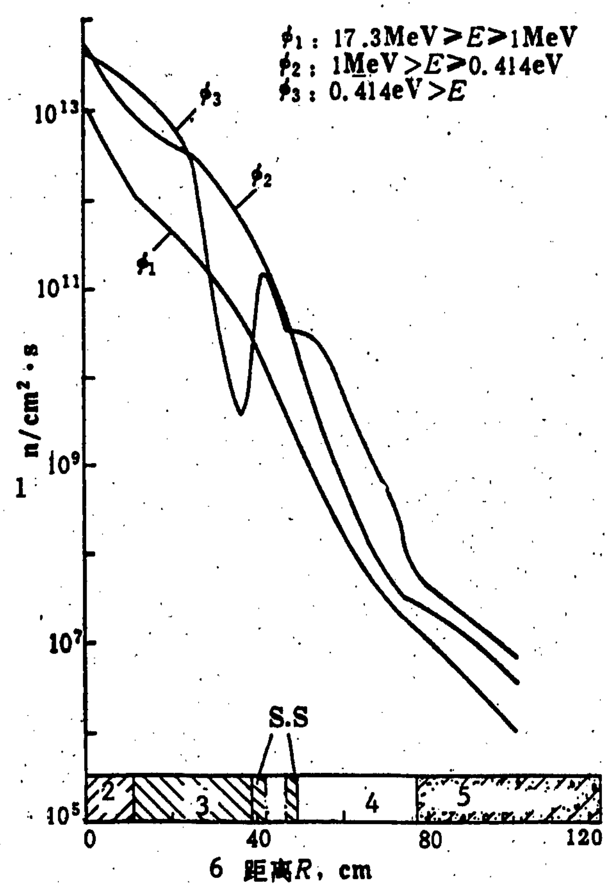


Figure 2. Radial Distribution of Neutron Flux

Key: 1. Neutron flux, n/cm^2 x s; 2. Beryllium; 3. Aluminum; 4. Water; 5. Concrete; 6. Distance R, cm

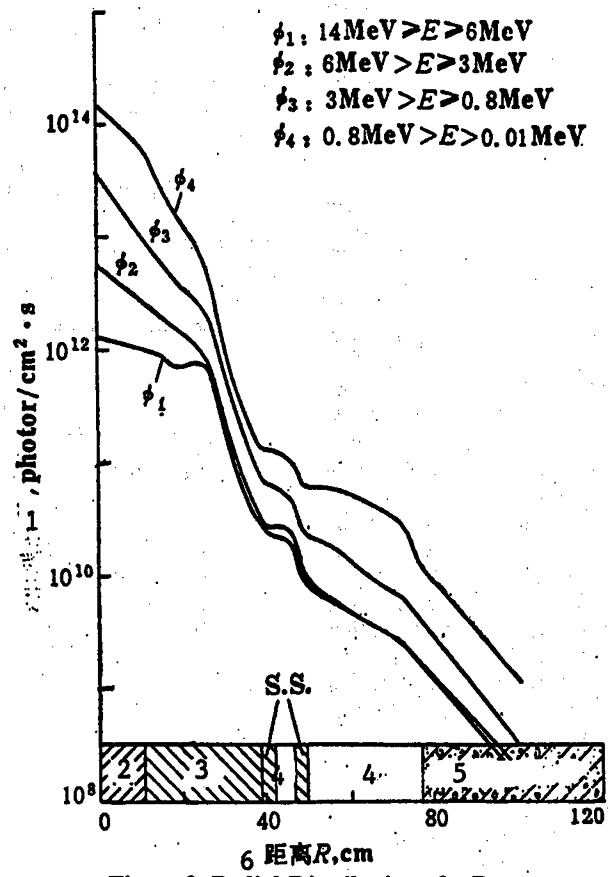


Figure 3. Radial Distribution of γ Rays

Key: 1. γ Rays, photon/cm² x s; 2. Beryllium; 3. Aluminum; 4. Water; 5. Concrete; 6. Distance R, cm

Table 3. Comparison of the Results Computed for the Original Design and Revised Design

Program	γ incident flux on the inner surface of the concrete, MeV/cm² x s	Thermal neutron flux in the monocrystalline silicon irradiation ducts, n/cm² x s	Thermal neutron flux in ionization chamber duct A, n/cm² x s
Original design	2.051×10^{11}	7.78×10^{12}	2.811×10^{10}
Revised design	2.854×10^{10}	7.44×10^{12}	2.213×10^{10}

VII. Concrete Shield Temperature Field Computations

During the process of being weakened and absorbed inside the concrete shield, the neutrons and γ rays transfer a portion of their energy to the concrete which heats the concrete and raises its temperature substantially. If the temperature is too high, it may cause a serious loss of crystallized water and can even result in fracturing. Moreover, if the temperature gradient is too great it can result in increased stress and an uneven stress distribution that can cause the concrete to crack. The after-effects could be destruction of the integrity of the concrete shield and a reduction in the shielding performance of the concrete.

The internal heat source in the concrete is mainly due to γ rays. Because the neutron flux is two quantum grades lower than the γ ray flux, neutron heating can be ignored and is not taken into consideration.

The temperature field computations were made using the one-dimensional shield temperature field program STFO. The internal heat source from γ ray heat release was computed with the ANISN program. The following assumptions were made regarding the temperature field:

1. Because the stainless steel sleeve cannot be linked tightly with the concrete, to make a conservative estimate it was assumed that there was a 0.1 cm air gap between the steel sleeve and the concrete;

2. The outside of the concrete is in close contact with loess rock strata, so a 50 cm loess rock strata outside the concrete was taken into consideration in the computations;

3. The inner boundary is a category 3 boundary condition and the fluid medium is water with a flow speed of 0.5 m/s and an average temperature of 45°C;

4. The outer boundary is assumed to be a fixed boundary and the temperature is assumed to be 15°C.

Figure 4 illustrates the temperature variations inside the concrete with an internal heat source and without an internal heat source. The maximum temperature inside the concrete obtained in the calculations is 49.16°C at a distance of 16.1 cm from the surface of the inner wall. The maximum temperature rise is 5.13°C and the maximum temperature gradient is 40°C/m. This result satisfies the stipulations in EJ-317.

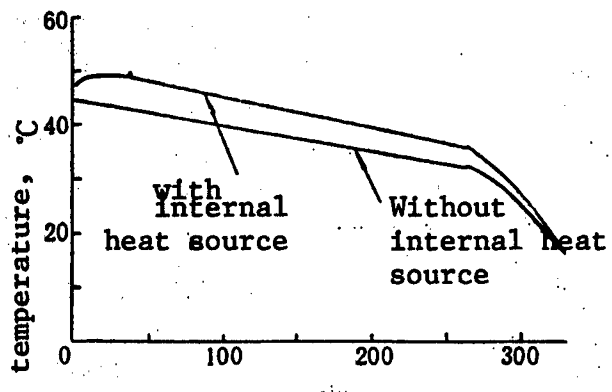


Figure 4. Radial Distribution of Temperature Inside Concrete

To explain the reliability of the temperature field computations, the authors used the internal heat source distribution inside the concrete plotted for the HFETR and the STFO program to do computations and compared them with the actual measured values (see Table 4). The computed results and measured results conformed rather well.

Table 4. Comparison of Computed Results and Actual Measured Values for the HFETR Concrete Shield Temperature Field

Radius, cm	Computed value, °C	Measured value, °C
114.5	58.3	59.15
125.5	59.01	59.4
130.5	57.46	57.7
145.5	52.85	53.18
175.5	42.40	47.45

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Accident Analysis for 5MW Low Power Reactor

926B0125I Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 pp 61-64, 74

[Article by Yu Junchong [0060 0193 1504], Tang Xueren [0781 1331 0088], Wang Defu [3769 1795 4395], and Wang Suhui [3769 4790 1979] of the China Nuclear Power Research and Design Academy, Chengdu: "5MW Low Power Reactor Accident Analysis"; manuscript received 22 February 1992]

[Text] Abstract: This article analyzes two types of serious accidents in the 5MW low power reactor (5MW LPR): rapid reactivity input accidents and loss of coolant accidents. The results show that if the reactor can be immediately shut down, no other measures are necessary to ensure the safety of the reactor under these two types of accident working conditions.

Key terms: 5MW low power reactor, reactivity accident, loss of coolant accident, residual heat.

I. Outline

Based on stipulations in HAF1001 and actual conditions in the 5MW LPR, we did a detailed analysis of its reactivity accidents, loss of coolant accidents, loss of flow accidents, loss of secondary coolant accidents, and pipeline plugging accidents. The results of the analysis indicate that the reactor would be safe under all the accident conditions described above. This article focuses on a description of the two most dangerous types of accidents among them: rapid reactivity input accidents and loss of coolant accidents [LOCA].

II. Initial Conditions

The initial conditions for accident analysis for the 5MW LPR are:

Initial reactor power: 103 percent of rated power, that is 5.15MW; power protection set value: 120 percent of initial power; water pool water level protection set value: alarm when the water level drops 0.10 m, reactor shutdown when the water level drops 0.50 m; control rod operation delay time: 0.05 s (0.4 s used in the calculations); control rod drop speed: insertion to the bottom within 1.0 s; total coolant flow rate: 620 t/h; average coolant flow rate in one box of fuel assemblies: 8.348 t/h; main pump rotational inertia:

0.26 kg x m^2; reactor inlet coolant temperature: 35°C (40.0°C used in the calculations); power inhomogeneity coefficient: $K_{xy} = 1.84$, $K_l = 1.10$, $K_z = 1.45$; heat flow density engineering heat channel factor: 1.03; enthalpy rise engineering heat channel factor: 1.08; moderator temperature coefficient: -1.60 x $10^{-4}\Delta K/K/°C$; during the reactor shutdown, it is assumed that a maximum value of one control rod is seized up outside of the core and that the total negative reactivity that can be input into the core is 14.75 elements.

III. Computing Programs and Models

Reactivity accidents were analyzed using the ADPARET program. This is a special program used for analytical research on reactor reactivity accidents. It uses a model that intercouples thermotechnical hydraulics with neutron dynamics and permits the core to be divided into four independent computation regions so that each region can independently input geometric, thermotechnical hydraulics, and dynamics parameters. A one-dimensional heat conduction and momentum integration model was used for the thermotechnical hydraulics and a point reactor model was used for the nuclear dynamics. The computation results from the programs conformed very well with the results of large reactor experiments.

WAP-2 was used for LOCA analysis. This is a special LOCA analysis program for pool-type reactors that is capable of describing eruption, siphoning, flow inversion, stagnant flow, natural circulation, exposure of the core to air, and several other types of thermotechnical hydraulics processes. Comparisons with the results of computations made using relevant programs in foreign countries indicate that the computed results with this program tend to be conservative and favor safety.

IV. Analysis Standards

During the accident process, it must be guaranteed that the integrity of the fuel elements is not damaged. The actual requirements are:

1. During the accident process, the core cannot be allowed to generate Departure from Nucleate Boiling (DNB). The Bernath relational expression was used to compute the Critical Heat Flow (CHF) density, and the DNBR should be greater than 1.20.

2. During the accident process, the temperature of the fuel element cladding should not exceed the permissible temperature under the ultimate heat stress.

V. Rapid Reactivity Input Accident

To determine the limit of rapid reactivity input that this reactor can withstand, it was assumed that the reactor can operate stably at 103 percent of rated power. When a loss of control accident involving the removal of a control rod from the core suddenly occurs, rather substantial reactivity can be input into the reactor within 1 second (for example 1 element/second, 2 elements/second, 3 elements/second,...), causing an abrupt rise in reactor power until the point where the action of the power protection system is initiated, thereby shutting down the reactor.

Analysis of the calculations indicates that if the reactor power protection system and control rod drive mechanism can operate normally, this reactor can withstand a 2 elements/second reactivity accident. The reactor power will reach the protection set value 0.1001 seconds after the accident and downward insertion of the control rods begins at 0.5001 seconds. At this time, the peak reactor power reaches 34.25MW, the minimum DNBR is 6.60, and the highest cladding temperature is 126.0°C, so the reactor is still very safe. See Table 1 and Figure 1 for the results in detail.

Table 1. Two Elements/Second Reactivity Accident Sequence Table

Time, seconds	Event occurring
0^-	Reactor is operating stably at 103 percent of power
0^+	Control rod loses control and is pulled out, normal reactivity input into reactor at the rate of 2 elements/second
0.10010	Reactor attains protection set value, reactor shutdown protection signal emitted
0.48755	Input of normal reactivity into the core attains the maximum value of 0.883 elements
0.50010	Downward insertion of control rods begins
0.51245	Reactor power reaches peak value of 34.25MW
0.6200	Fuel temperature reaches peak value of 128.5°C
	Cladding temperature reaches peak value of 126.0°C
	DNBR reaches minimum value of 6.60
	Reactor power drops to 4.2MW
0.79050	Heat channel outlet water temperature reaches peak value of 92.64°C
1.50010	Control rods inserted to bottom
	Reactor power drops to 0.75MW
	Heat channel outlet water temperature drops to 89.25°C
	Highest temperature of element cladding drops to 102.4°C
	Highest temperature of fuel drops to 104.2°C

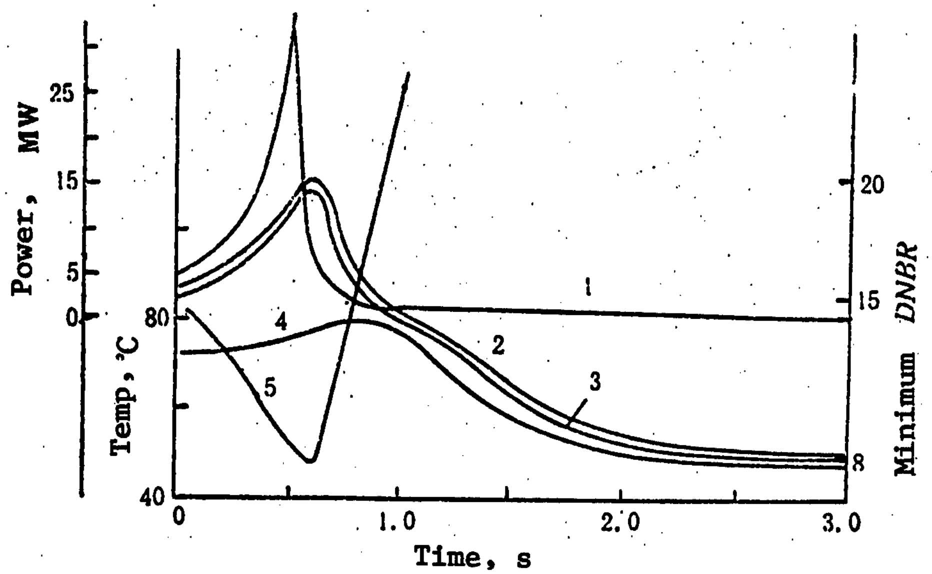

Figure 1. Radial Distribution of Temperature Inside Concrete

Key: 1. Reactor power; 2. Highest temperature of fuel kernels; 3. Highest temperature of cladding surface; 4. Heat channel outlet water temperature 5. Minimum DNBR

VI. Loss of Coolant Accident Analysis

The 5MW LPR is installed in an 8.5 x 2.2 x 8.9 m large water pool. An assumption was made that a loss of coolant accident occurs because of a rupture at both ends of a pipeline at the lowest position outside of the water pool in the φ325 x water intake pipe (at a standard height of 0.0). After the water intake pipe ruptures, the water pool receives no replenishment water of any kind, the main pumps prior to ceasing operation must pump water outside of the water pool, and the formation of a siphon at the site where the pipe has ruptured also causes water to flow out of the water pool, thereby resulting in a rapid drop in the water level in the water pool. When the water level drops 0.10 m, the water level monitoring system emits an alarm signal. When it drops 0.50 m, it emits a protective reactor shutdown signal.

To obtain conservative results, the following assumptions were made in the computations and analysis: 1) The time interval between the beginning of the LOCA and the reactor shutdown is no less than 60 seconds; 2) The main pumps do not stop until 60 seconds after reactor shutdown; 3) There is a minimum resistance in the pipe section where the siphon occurs; 4) After the main pumps stop operation, the auxiliary cooling system does not go into operation; 5) No water is replenished in the water pool.

The calculations indicate that a water level alarm signal would be emitted 4.3 seconds after the accident occurs, a reactor shutdown signal would be emitted at 21.84 seconds, the water level would drop to a 0.0 standard height at 314.0 seconds, the siphon would cease, and the water level would drop no more. At that time, there would still be a water layer 1.37 m deep at the top of the reactor, which means that a water intake pipeline rupture accident would not expose the core. During the entire accident process, the heat from the core would be removed by the natural circulation it had established itself to the water pool and the reactor would be very safe. See Table 2 and Figures 2 and 3 for the detailed results.

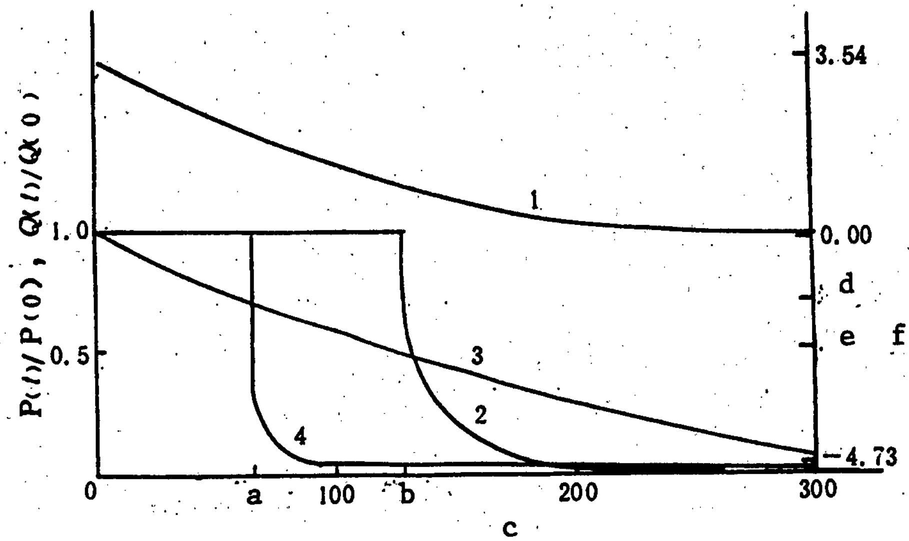

Figure 2. Changes in Relevant Parameters Inside Water Pool Over Time During Loss of Coolant Process
Key: a. Reactor shuts down; b. Pumps shut down; c. Time, s; d. Core inlet; e. Core outlet; f. Water level in water pool, m; 1. Water level in water pool; 2. Main pump flow rate; 3. Siphon flow rate; 4. Reactor power

Table 2. Loss of Coolant Accident Sequence Table

Time, seconds	Event occurring
O⁻	Reactor is operating stably at 103 percent of rated power
O⁺	Rupturing occurs at both ends of the water pool water intake pipe at a standard height of 0.0 m
4.3	Water level drops 0.10 m, alarm signal emitted
21.84	Water level drops 0.50 m, reactor shutdown signal emitted
60.0	Reactor shuts down
120.0	Main pumps shut down
169.5	Core coolant flow inversion begins, establishment of natural circulation begins. During the process of establishing natural circulation, the heat channel water temperature and cladding wall temperature reach their first peak values at 103°C and 124.3°C, respectively, minimum DNBR is greater than the stable state value
314.0	Water pool water level drops to 0.0 standard height
≈ 1100.0	Heat channel outlet water temperature and cladding wall temperatures reach their second peak values at 80.5°C and 110.0°C, respectively. Afterwards, they gradually drop.

Because it was assumed that the auxiliary cooling system did not go into operation, all of the residual heat removed from the core was released into the water pool. Would this result in heating, boiling, and vaporization of the water pool, thereby exposing the core to the air and causing it to burn up? The calculations indicate that the water 1.37 m deep at the top of the core weighs at least 25 tons. To heat this water to saturation, without including heat dissipation losses, it would take over 6.0 x 10^6 kJ of heat, which would mean that it would take the release of residual heat for at least 12 hours. This means that there would be sufficient time to repair the rupture pipe or find other ways to replenish the water in the water pool. Thus, there is no risk of the core being exposed due to evaporation of the water in the pool from the release of residual heat.

After the main pumps cease operation and with the auxiliary cooling system not going into operation, if the core did not establish its own natural circulation, could the safety of the reactor be guaranteed? The calculations and analysis indicate that after the main pumps stop operation, the inertial flow rate would also be maintained for 60 seconds. When the coolant reached a rest state in the core, it would be heated very quickly to saturation and part of it would be vaporized, causing a heat and mass exchange between the vapor and 1.37 m deep water layer at the top of the reactor and maintaining the core coolant in a mixed steam-water saturated state. Under this state, the highest temperature of the surface of the fuel element cladding could reach 244°C. The results of thermal stress experiments indicate that damage to the cladding would not occur until the

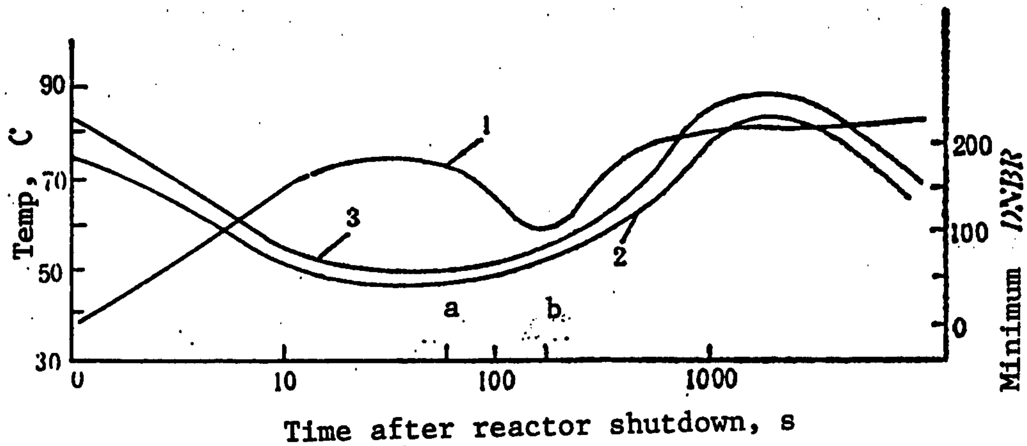

Figure 3. Thermotechnical Characteristics of Heat Channel During Loss of Coolant Process
Key: a. Pumps shut down; b. Inversion; 1. Minimum DNBR; 2. Heat channel outlet temperature; 3. Highest wall temperature of fuel elements

fuel element cladding had undergone 2,800 heat shocks at 250°C. It is apparent that during the LOCA process, the fuel elements would still not be damaged even if the reactor failed to establish natural circulation.

VII. Conclusion

The analysis above indicates that the 5MW LPR has very good accident safety. Even in the more serious type of accidents—a 2 elements/s positive reactivity input accident and water pool loss of coolant accident—if the control and protection systems and control rod drive mechanisms merely function normally, the reactor would always be safe without having to adopt other measures.

Physical Start-Up of 5MW Low Power Reactor

926B0125J Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 pp 65-68

[Article by Yu Weide [0060 4850 1795] and Li Zhidong [2621 1807 2767] of the China Nuclear Power Research and Design Academy, Chengdu: "5MW Low Power Reactor Physical Startup"; manuscript received 21 February 1992]

[Text] Abstract: This article describes the first loading and first criticality test of the 5MW low power reactor (5MW LPR). Because this reactor uses consumed fuel elements and beryllium as a reflecting layer, photoneutrons become a powerful internal neutron source whose intensity increases as the amount of fuel elements loaded is increased. Thus, during loading there was substantial variation in the results of element method extrapolation, but the overall loading process was safe. At subcriticality, a relative efficiency curve for the control rods was prepared and then the control rods were lifted, count extrapolation was carried out, and first criticality was achieved.

Key terms: 5MW low power reactor, loading, physical startup, count extrapolation, first criticality.

I. Introduction

Physical startup of the 5MW LPR and the attainment of first criticality are indicators of the reactor's physical operability. The important parameters obtained from criticality tests provide a foundation for debugging during the increased power stage and provide necessary data on safe reactor operation. Moreover, the physical startup process is a comprehensive test of the reactor's control and protection systems that examines and confirms the reliability and response capabilities of the systems and instruments.

During physical startup, because processing of certain core components had not been completed, they could not be placed in the reactor, so the first criticality loading was somewhat different from the actual loading of the 5MW LPR. See Figure 2 in the article "The 5MW Low Power Reactor" [ref 926B0125B above] in this special edition of HE DONGLI GONGCHENG for the first criticality loading of the 5MW LPR. The core had 32 boxes of fuel elements and 10 control rods, the reflecting layers had 44 beryllium blocks and 153 aluminum blocks, no core components were loaded in the central K_{11}, a water cavity was substituted for the molybdenum-technetium irradiation ducts, the aluminum reflecting layer was configured with four monocrystalline silicon irradiation ducts, and to remove the water cavity left after replacement of the aluminum blocks, the 48 stainless steel blocks around the outside that function as heat shields were not placed into the reactor and were replaced with a water cavity. In the aluminum reflecting

layer K_3, temporary aluminum guide tubes were inserted in positions B_4, E_{14}, U_{14}, and T_{17}. In them, a startup neutron source was placed in K_{11} and detectors used for startup were placed in the others.

II. Special Problems in Startup and Pre-Startup Preparations

The 5MW LPR is a swimming pool-type reactor with beryllium as a reflecting layer and light water as a moderator that uses fuel elements that have been used in the high-flux reactor. The average burnup of a box of the fuel elements that were used in the high-flux reactor is 38 to 40 percent. Even though they had cooled for 1 to 2 years after being unloaded, they still had rather powerful decay γ rays and the γ rays interacted with the beryllium in the reflecting layer to form a photoneutron source. With just one box of fuel elements in the reactor, the intensity of the photoneutrons could exceed the neutron source used in startup and cause the reactor to have a rather high background neutron flux at deep sub-criticality. While this might not require the use of a startup neutron source, the neutron flux level when the reactor approaches criticality might be too high, which would result in the neutron fission chamber inside the reactor barrel used for startup to exceed the count limits, so during the startup process the fission chamber had to be moved to an outer layer of detection guide tubes. At the same time, the existence of photoneutrons also presented difficulties for element method extrapolation during the loading process. Because the intensity of the photoneutrons increases as the amount of fuel that is loaded increases, sub-criticality multiplication based on a fixed external neutron source and point reactor model may result in greater inaccuracy. This is particularly true under deep sub-criticality conditions when the increase in the neutron count resulting from neutron multiplication might be far smaller than the increase in the neutron count caused by photoneutrons. In this type of situation, the extrapolated values may change very little and result in an inability to add fuel elements to the core based on safety limits. Actually, the extrapolated values in this type of situation are not true. The minimum element load can be determined based on empirical data from the high-flux reactor to ensure the critical safety of the reactor during the loading process.

Given these special problems, a great deal of preparatory work was done prior to startup, including physical computations, neutron source problems, deployment of startup counting devices and detectors, the possible occurrence of accidental situations during the loading and sub-criticality extrapolation processes, and so on. After repeated and conscientious analytical research, the "Low Power Reactor Loading and Sub-Criticality Test Procedures" were compiled and first criticality loading charts were drawn. The loading and first criticality were carried out strictly according to the "Procedures" and loading diagrams.

To do count monitoring and extrapolation during the processes of loading and raising the control rods to move closer to criticality, we prepared two sets of neutron counting devices. To ensure critical safety during the startup process, we prepared two sets of pulse cycle protection and two sets of small power protection devices, and we added two sets of power protection devices. To ensure the reliability of the count monitoring at deep sub-criticality and guarantee the safety of the entire startup process, we installed four temporary detector guide tubes inside the barrels of the reactor. They included counting device probes placed at positions E_{14} and T_{17} (in the neutron fission chamber; they also provide pulse cycle protection and small power protection signals), and we placed power protection device probes at positions B_4 and U_{14} (in the neutron ionization chamber). When the inner layer fission chamber count exceeds the limits, the counting device can be easily switched over to the fission chamber in the outer layer of detector guide tubes to ensure the continuity of count monitoring.

Prior to physical startup, debugging and testing of these startup devices and detectors was done and tests were done of the control rod raising and lowering speeds. All the results of debugging and tests satisfied the design requirements.

Prior to formal loading, besides the 32 boxes of fuel elements and the eight boxes of reflecting layer beryllium blocks, all of the other components inside the reactor were in place.

III. Loading and First Criticality Test and Results

The loading and first criticality test in the 5MW LPR was carried out under the supervision of the National Nuclear Safety Administration and Beijing Assessment Center. It began at 0800 hours on 2 February 1991 and continued until the safe attainment of first criticality at 1842 hours on 2 February 1991.

Prior to loading, we first raised two safety rods and used the startup counting devices to measure neutron background data, after which the startup neutron source was inserted into the temporary guide tube at position K_3 that had been prepared beforehand. The entire loading process was carried out in three steps: In the first step the initial loading was completed by adding in sequence 16 boxes of fuel elements at locations J_{10}, L_{12}, K_{10}, K_{12} L_{10}, J_{12}, K_9, K_{13}, J_9, L_{13}, M_{10}, I_{12}, I_8, M_{14}, L_9, and J_{13} and the neutron count was recorded and the count monitored after each box of elements was loaded. In the second step the pre-loading was completed by adding 10 boxes of fuel elements in sequence at locations M_{12}, I_{10}, J_8, L_{14}, K_8, K_{14}, N_{12}, H_{10}, N_{13}, and H_9 and adding eight boxes of beryllium blocks at locations F_6, Q_{16}, Q_{11}, F_{11}, I_6, M_{16}, M_8, and I_{14}, the neutron count was recorded and count extrapolations were done after each box of elements or beryllium was loaded. In the third step, the rated loading was completed by adding six boxes of fuel elements in sequence at locations L_{11}, J_{11}, H_8, N_{14}, N_{11}, and H_{11}. When each box of elements was loaded, a neutron count

was recorded and a count extrapolation was made to derive the critical element box count and determine the number of elements to be added the next time. Table 1 lists the results of element method extrapolation during the process of completing the rated loading and Figure 1 is a plot of the element method extrapolation curve.

Table 1. Loading Extrapolation Count

Sequence	Location inside reactor	Number of elements in reactor, boxes	Neutron counting rate, s^{-1}		Extrapolated critical element loading count, boxes	
			No 1 count loaded amount	No 2 count loaded amount	No 1 count loaded amount	No 2 count loaded amount
0	-	26	2544	8630	-	-
1	L_{11}	27	2750	9681	39.3	35.2
2	J_{11}	28	3146	10596	34.9	38.7
3	H_8	29	3462	11546	39.0	40.0
4	N_{14}	30	4048	16772	35.9	32.2
5	N_{11}	31	4448	20421	41.1	35.6
6	H_{11}	32	5856	21991	35.2	45.0

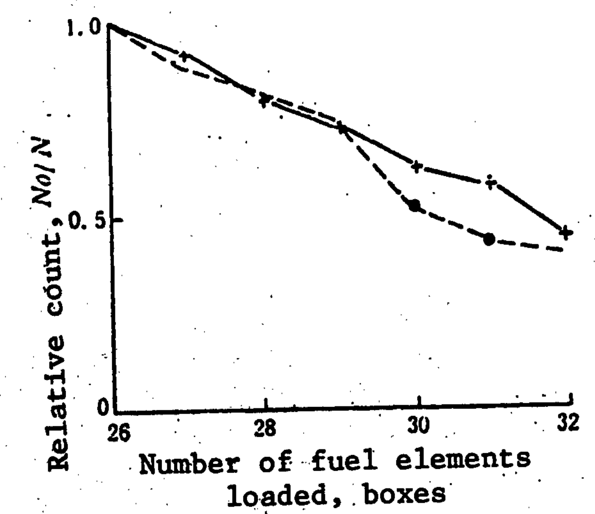

Figure 1. Element Method Extrapolation Curve

When the loading was completed, the safety rods were lowered. Extrapolation was carried out for the neutron count based on two safety rods being fully raised and fully lowered. It was estimated that in a situation in which two safety rods were raised, the reactor's subcriticality would be equal to 1.25 times the equivalent of two safety rods. For this reason, there was a sufficient subcriticality margin and the rods could be rasied to plot the efficiency curve of the control rods. At this time, the two sets of counting devices already had sufficiently high neutron counts and the neutron source was removed.

After raising the two safety rods, the No 6 compensation rod was raised by sections. When raised by each section, the neutron count was recorded and the relative efficiency curve was plotted for that control rod. Figure 2 shows the relative efficiency curve for the No 6 compensation rod.

During the first criticality, besides the two safety rods, the Nos 1 and 2 compensation rods and Nos 3 and 6 compensation rods were raised (see Table 2 for the control rod numbers and their position inside the reactor). See Table 3 for the extrapolated criticality and the steps in the transition, the elevation to which each control rod was raised, and the extrapolated data for the counting devices. When the Nos 1 and 2 compensation rods were at 50 cm and the Nos 3 and 6 compensation rods were at a height of 25 cm, the extrapolation results from the two sets of counting devices were identical, $K_{eff} > 0.998$. Next, the No 3 compensation rod was raised to 32.2 cm and the measured reactor multiplication cycle was 92 s.

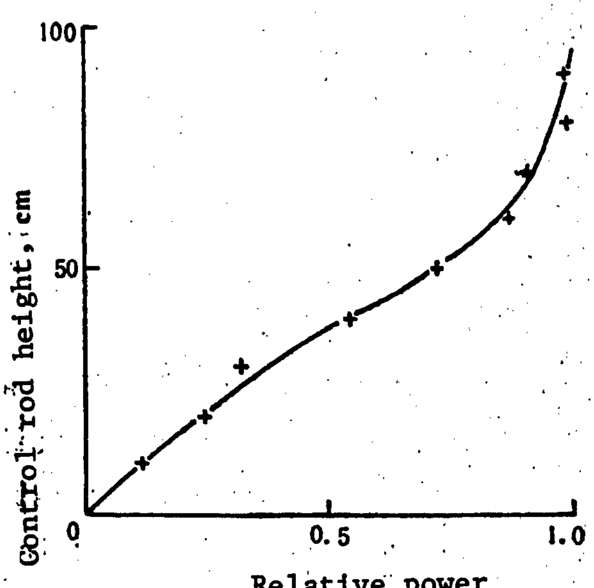

Figure 2. Number 6 Compensation Rod Effectiveness Curve

Table 2. 5MW LPR Control Rod Numbering

Name	Number	Location in reactor
No 1 safety rod	1AB	I_9
No 2 safety rod	2AB	M_{13}
No 1 compensation rod	1SB	M_{11}
No 2 compensation rod	2SB	I_{11}
No 3 compensation rod	3SB	I_{13}
No 6 compensation rod	6SB	M_9
No 4 compensation rod	4SB	M_{15}
No 7 compensation rod	7SB	I_7
No 1 regulation rod	1ZB	P_{13}
No 2 regulation rod	2ZB	G_9

Table 3. First Criticality Rod Lifting Extrapolation and Results

Sequence	Height to which control rods raised, cm				Neutron counting rate, s^{-1}		Extrapolated subcriticality, β_{eff}		Notes
	1.2AB	1.2SB	3SB	6SB	No 1 count	No 2 count	No 1 count	No 2 count	
0	100				5896	14293	-	-	
1		25			9315	17696	5.39	13.13	
2		30			10996	19718	3.50	5.51	
3		35			13264	22694	3.05	4.17	
4		40			16201	27270	2.84	3.12	
5		50			29225	47248	1.57	1.72	
6			15		46235	56163	0.62	1.91	
7				10	56181	64327	1.02	1.51	
8				20	77901	84330	0.72	0.90	
9	100	50	15	20	446	314	-	-	
10			25		815	579	0.36	0.36	No 1 count connected to 1QDS, No 2 count connected to 2QDS
11				25	1696	1267	0.15	0.13	$k_{eff} > 0.998$
12	100	50	32.2	25	-	-	-	-	Multiplication cycle = 92 s

IV. Conclusion

The loading and first criticality of the 5MW LPR were completed satisfactorily and passed witness by the National Nuclear Safety Administration. The results of the criticality tests provided the prerequisite conditions and necessary parameters for debugging in the power phase.

After completion of the first criticality test, the stainless steel blocks were placed in the 48 water cavity locations around the outside of the reactor core and criticality was restored on 15 March 1991. The critical rod positions were slightly lower than subcriticality. The critical rod positions were: 1, 2AB fully raised; 1SB 50.1 cm, 2SB 49.9 cm, 3SB 26.9 cm, and 6SB 25.0 cm.

72-Hour Test at Full Power for 5MW Low Power Reactor

926B0125K Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 pp 75-80

[Article by Li Maoyuan [2621 5399 0337], Pu Yunde [2528 0061 1795], Liang Guangyuan [2733 0342 6678], Zhang Keqiang [1728 0344 1730], Zhang Qin [1728 0530], and Zhang Liangwan [1728 5328 8001] of the China Nuclear Power Research and Design Academy, Chengdu: "5MW Low Power Reactor Physical Startup"; initial manuscript received 9 March 1992, revised manuscript received 30 March 1992]

[Text] Abstract: The article describes the overall continuous full-power operation conditions of the 5MW low

power reactor (5MW LPR) and operating experience. The results of the test indicate that performance of the reactor body and all technical systems attained design requirements and can operate normally, and that this reactor has a rather substantial thermotechnical safety margin.

Key terms: 5MW low power reactor, full-power operation, safety margin.

I. Introduction

Based on the related requirements of the National Nuclear Safety Administration, after completing all tests of the 5MW LPR in stage C, a continuous operation test for 72 hours at full power was conducted from 30 July to 2 August 1991. It confirmed that this reactor performed excellently and attained design requirements. It also accumulated experience and data for future safe operation and enabled operating and maintenance personnel to gain an understanding of the operating characteristics and methods of this reactor.

II. Objectives and Conditions for the Operation Test

The objectives of the operation test were to examine whether or not the construction conformed to design requirements and whether or not the systems and equipment could perform their expected functions, to examine and accept whether or not the reactor could complete the various types of expected functional indices, to gain baseline data for safe operation and examine and accept whether or not the basic conditions used in the safety analysis were appropriate, to inspect the suitability of the operation regulations and management system, and to train and temper the first group of operating and maintenance personnel to enable them to understand and become familiar with the operation characteristics and methods of this reactor.

Based on the working conditions required for the operation test, the functions of these auxiliary systems were tested and they were placed into operation: the power supply system, ventilation system, communications system, dose monitoring system, secondary water system, primary cooling system, special drainage system, waste water treatment system, purification system, water replenishment system, residual heat removal system, and thermotechnical measurement system.

Besides the things outlined above, the following conditions also had to be prepared:

1. Operating and maintenance personnel were already familiar with and understood all operation regulations and management systems;

2. An ability to correctly analyze and deal with problems that might appear during the operation test and the measures that should be adopted;

3. Clear duties and division of labor of the personnel participating in the test;

4. Compilation of the "High-Power Reactor Startup Preparations Card", "Reactor Startup Operations Card", "Auxiliary System and Primary Cooling System Operations Card", "Reactor Shutdown and System Shutdown Operations Card", and the documents required for the test;

5. Main control room operating personnel must have advanced operating personnel and operating personnel licenses.

III. Primary Parameters Set Values and Actual Operation Values

To protect the safety of the reactor, every system of the low power reactor has set values for the automatic alarms and autonomous region reactor shutdown (see Table 1). The operation limits of the low power reactor stipulated these set values during the operation test. If a situation appears that endangers the safety of the reactor, the reactor must be shut down either manually or automatically.

Table 1. Set Values for Major Parameters for the 5MW LPR and Actual Operation Values

Parameter	Warning value	Accident value	Actual operation value	Rated working conditions design value
Thermal power (high), MW	5.5MW	6.0MW	5MW	5MW
No 1 nuclear power (high), %	110%	120%	100%	-
No 2 nuclear power (high), %	110%	120%	100%	-
No 3 nuclear power (high), %	110%	120%	100%	-
No 1 cycle (short), s	38 (e times)	22 (e times)	> 50 (e times)	-
No 2 cycle (short), s	38 (e times)	22 (e times)	> 50 (e times)	-
Primary water flow rate (low), t/h	560	505	625	620

Table 1. Set Values for Major Parameters for the 5MW LPR and Actual Operation Values (Continued)

Parameter	Warning value	Accident value	Actual operation value	Rated working conditions design value
Secondary water flow rate (low), t/h	810	-	920	1000
Reactor pool liquid level (low), m	3.44	3.05	3.50	3.54
Reactor pool liquid level (high), m	3.64	-	3.50	-
Water replenishment tank water level (low), m	1.75	-	2.01	-
Water replenishment tank water level (high), m	2.28	-	2.05	-
Purification inlet temperature, °C	44	-	27	-
Hot box of elements outlet water temperature (high), °C	L_{12}: 70°C	-	51	-
	N_{14}: 70°C	-	41	-
	J_{10}: 70°C	-	53	-
	H_8: 70°C	-	45.8	-
External power source loss of power	-	< 75% rated value	≈380V	-
Primary water outlet temperature, °C	-	-	34	47
Secondary water outlet temperature, °C	-	-	27	40

During the operation test, the primary loop water quality stayed within the stipulated standards (see Table 2), which is an indication that the purification system satisfied the design requirements.

Table 2. Water Quality Parameters During the 5MW LPR Operation Test*

Item	Time of sample		
	2108 hours, 30 July 1991	0900 hours, 31 July 1991	0945 hours, 2 August 1991
pH value	6.0	6.0	6.1
Specific resistance, O/cm	106.7×10^4	103.5×10^4	153.9×10^4
Total solids, mg/L	0.3	0.3	0.3
Cl^{-1}, mg/L	<0.1	<0.1	<0.1
Cu^{++}, mg/L	<0.01	<0.01	<0.01
Pb^{++}	<0.01	<0.01	<0.01
^{144}Ce, kBq/L	10.21	13.7	36.8
^{141}Ce, kBq/L	0.464	0.500	-
^{131}I, kBq/L	3.22	5.37	10.24
^{133}I, kBq/L	6.66	16.4	7.42
^{137}Cs, kBq/L	2.40	2.81	10.84
^{24}Na, kBq/L	15.6	65	53.2
Total γ, kBq/L	8112.7	5850.6	7025.3

*Reactor power during test was 5MW, sampling location was P sample 06.

To carry out radiation monitoring in the low-power reactor, the dose monitoring system was placed into operation based on the relevant instructions during the 72-hour full-power operation test. The measurement results are given in Tables 3, 4, and 5.

Table 3. Technical Workshop γ Field Measurement Results During 5MW LPR Operation Test

Time of measurement	γ field measurement point count, x 2.58 x 10^{-16} C/kg x s							
	Measurement point							
	104-1	104-2	109-1	109-2	108	116	203	207
Background	0.02	0.015	0.025	0.015	3.20	0.15	0.05	0.05
1800 hours, 30 July 1991	1.85	12	170	300	260	0.38	0.05	0.05
0600 hours, 31 July 1991	2.50	12.5	170	290	1150	0.55	0.05	0.05
1800 hours, 31 July 1991	2.50	13	175	290	1550	0.55	0.05	0.05
0600 hours, 1 August 1991	3.00	12	175	300	1900	0.25	0.05	0.05
1200 hours, 1 August 1991	2.70	13	220	300	2350	0.25	0.05	0.05
1800 hours, 1 August 1991	2.70	13	175	300	2550	0.23	0.05	0.05
0600 hours, 2 August 1991	3.2	13	175	300	2650	0.22	0.05	0.05
1200 hours, 2 August 1991	3.2	18	175	300	2650	0.22	0.05	0.05
17 hours after reactor shutdown	0.12	0.5	5	5	420	0.04	-	-

Table 4. Average Aerosol Concentration During 5MW LPR Operation Test, Bq/L

Sample point/Time of sample	20 minutes after sampling	7 times after sampling
Reactor pool building	3.33 x 10^{-3}	4.4 x 10^{-6}
Primary loop room	3.23 x 10^{-3}	4.9 x 10^{-6}
Damage detection room	5.88 x 10^{-3}	3.4 x 10^{-6}

Table 5. Radioactive Gas Concentration in Technical Workshop During 5MW LPR Operation Test, x 3.7 Bq/L

Time of measurement	Reactor pool building	Primary water main loop room	Damage detection room
Background	1.2	1.2	1.2
1800 hours, 30 July 1991	1.2	1.1	1.3
0600 hours, 31 July 1991	1.3	1.0	1.2
1200 hours, 31 July 1991	1.2	1.5	1.4
1800 hours, 31 July 1991	1.4	1.2	1.4
0600 hours, 1 August 1991	1.2	1.1	1.3
1200 hours, 1 August 1991	1.2	1.4	1.5
1800 hours, 1 August 1991	1.3	1.3	1.4
0600 hours, 2 August 1991	1.2	1.4	1.4
1200 hours, 2 August 1991	1.3	1.3	1.3
17 hours after reactor shutdown	1.2	1.3	1.3

IV. Test Results and Discussion

A. On assessing xenon poisoning

Because the 5MW LPR uses elements unloaded from the high-flux reactor, the average burnup of the elements placed into the reactor is about 40 percent. The positive reactivity effects and temperature effects of the ^{149}Sm in the burned-up elements posed substantial difficulties for measuring xenon poisoning, so we were only able to make a comprehensive assessment. After 40.5 hours of operation, the comprehensive reactivity of the reactor attained equilibrium (xenon poisoning, temperature, burnup, and ^{149}Sm positive reactivity). Overall reactivity losses were about 5 β_{eff}. The positive reactivity of the ^{149}Sm was released after 42.5 hours, which was manifested at automatic rod re-insertion at a significant speed (greater than the poisoning, burnup, etc.). This characteristic of the low-power reactor attracted the attention of operating personnel.

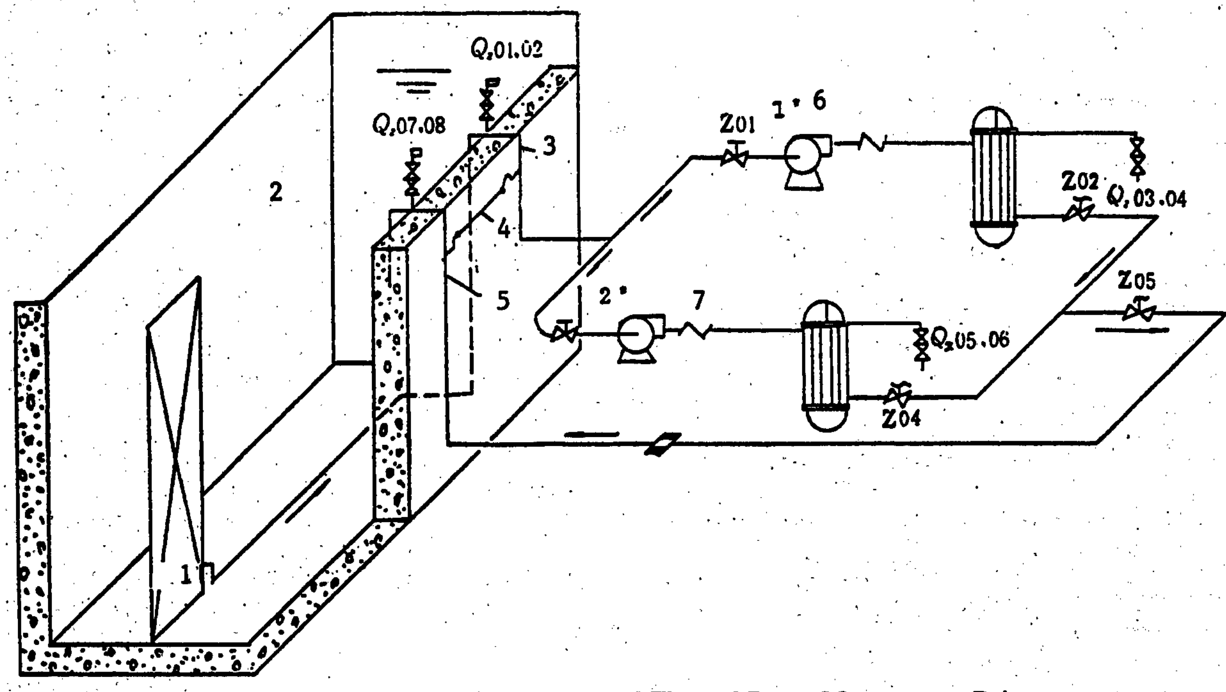

Figure 1. 5MW LPR Primary Loop and Thermal Power Measurement Points
Key: 1. Core; 2. Reactor water pool; 3. Water pool outlet; 4. Temperature measurement point; 5. Water pool inlet; 6. Number 1 main pump; 7. Number 2 main pump

B. Effect of rod lattice height on nuclear measurement instruments

During the 72-hour full-power operation test, under conditions of no variation in reactor power, when the No 1 automatic rod position was raised high, there was a tendency for the No 2 power measurement reading count to increase. The was because the configurations of the No 2 power measurement detector and the No 1 automatic rod in the core (see Figure 2 in the article "5MW Low Power Reactor" [ref 926B0125B above] in this special issue of HE DONGLI GONGCHENG) are on the same side, so raising the manual rod position also caused an increase in the No 2 power measurement. For this reason, when looking at the nuclear power numerical value, attention must be given to the effect of the height of the rod lattice. Moreover, attention also must be given to whether or not the later period of burnup causes ion saturation so that immediate readjustments can be made as necessary in the height of the ionization chamber.

C. Thermotechnical measurements

Because this reactor is a swimming pool reactor, the water pool itself has a very large heat capacity. Moreover, the reactor's thermal power measurement points are placed on the inlet and outlet mother pipes of the water pool instead of the inlets and outlets of the core, as shown in Figure 1.

Figure 1 shows that the thermal power measured at the measurement point is the output power of the entire water pool. Prior to attaining thermal equilibrium, it should include all of the heat generated by the core as well as the overall sensible heat variation value of the water pool. After attaining thermal equilibrium, the variation in overall sensible heat in the water pool tends toward zero, and only then is the thermal power that is measured truly the power of the reactor core. In terms of actual operation conditions, at full-power working conditions, it takes about 1.5 to 2 hours to reach thermal equilibrium. Calculating based on a flow rate of 620 t/h and a system total water volume of about 160 m^3, there must be 6 to 8 heat exchanges between the entire primary water and secondary water before they finally attain thermal equilibrium. This is the greatest characteristic of the 5MW LPR, and this characteristic deserves the attention of operating personnel.

D. Thermotechnical safety margin

The operation test indicated that at operating conditions of a secondary water flow rate of 900 t/h and the hottest climatic conditions, the reactor's primary water inlet and outlet temperatures are lower than the design value 13°C and the hot box elements outlet water temperature is also lower than the design value 10 to 15°C. This shows that the 5MW LPR has a rather substantial thermotechnical safety margin.

E. Dose monitoring

The data monitored by the dose monitoring system listed in Tables 3 to 5 show that the doses outside the shielding

layers of all the technical workshops are the background level. Under full-power operation conditions, the γ radiation at the top of the reactor top cover plate and the top of the reactor outlet mother pipe concrete top plate is 1.032×10^{-8} Ci/kg/s. Because the monocrystalline silicon ducts at the top of the reactor do not have shielding plugs, the γ radiation is 5.16×10^{-8} Ci/kg. The radioactive gas and aerosol concentrations in the reactor pool building and primary loop system are both at background levels.

F. Element damage detection

The element damage detection system was placed into operation during the operation test, and we also carried out regularly scheduled water sample nuclide analysis. The results show no significant increase in the primary nuclides ^{137}Cs and ^{131}I in the fission products, which indicates no damage to the elements. This prepared the technical conditions for clarifying the background values for the damage detection system and additional tests of the system.

G. Other experiments

During the 72-hour full-power operation test, while ensuring the operation test, we also completed Mo-Tc isotope irradiation experiments. The results show that the maximum specific activity of the Mo-Tc isotopes could reach 620 mci, with a minimum of 422 mci. Calculations indicate that the Mo-Tc isotopes produced from 8 to 10 days' full-power operation at 5MW of this reactor could satisfy user requirements.

H. Comprehensive assessment

The 72-hour full-power operation test shows that all technical systems in the 5MW LPR operated normally and that the equipment functions and operability attained design requirements. The operation results also revealed that the thermotechnical design parameters of this reactor tended toward safety and had considerable latent potential.

V. Operating Experience

The reliability and safety of the reactor design is the most important foundation for long-term continuous operation, and the operability of all of the equipment and systems, the quality of the operating personnel, and effective management are the decisive conditions in guaranteeing safe reactor operation. The main experience gained from the 72-hour full-power operation test includes these points:

A. Strict operating discipline, include safe reactor operation in the legal system

The 5MW LPR is a reactor that was operated after passing safety assessment and debugging supervision by the National Nuclear Safety Administration. Various systems of relevant regulations were established in accordance with the stipulations in relevant laws and regulations. This basically eliminated operational behavior that lacked stipulations and foundations. On the other hand, carrying out education on discipline and regulations for operating personnel to improve the quality of operating personnel made regulations and all relevant systems the basis for operating personnel behavior, which prevented the occurrence of blind and baseless operational behavior and guaranteed safe reactor operation.

B. Strict observance of operation limits and conditions

The operation limits and conditions approved by the National Nuclear Safety Administration were an important foundation for safe reactor operation by the management unit. During operation, a group of operation limits and conditions that conform to reality are extremely important for safe operation. For operating managers and operating personnel, it is both a legal restriction and a type of guide to ensure that reactor operation working conditions conform to limits at all times.

C. Operational supervision and inspection tour supervision point inspection system

During the operation test, safety, quality assurance, and other departments implemented effective supervision and cooperated closely with operating personnel, which provided an important guarantee of safe operation. In addition, the physical and thermotechnical characteristics of the 5MW LPR indicated that this reactor has rather good inherent safety and that safe operation of the reactor can be guaranteed merely by ensuring that the operating equipment and important safety measures are normal and reliable. However, because operating personnel lacked operational experience, an inspection tour supervision point inspection system was formulated to ensure that the systems and equipment functioned normally to ensure safe operation.

VI. Conclusion

The 72-hour full-power operation test of the 5MW LPR was satisfactorily successful and indicated that this reactor had been formally completed. In the future, along with studying burnup of HFETR fuel elements, we will gradually carry out monocrystalline silicon irradiation doping, Mo-Tc ionization and gemstone irradiation and production, and explore the possibility of developing neutron activation analysis and indium-gallium-tin γ loops.

Inherent Safety Research on 5MW Low Power Reactor

926B0125L Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 pp 81-86

[Article by Wang Zhendong [3769 2182 2639], Zhang Liangwan [1728 5328 8001], Xia Guanghua [1115 0342

5478], Zhang Zicai [4545 1311 2088], Yu Junchong [0060 0193 1504], Pu Yunde [2528 0061 1795], and Li Maoyuan [2621 5399 0337] of the China Nuclear Power Research and Design Academy, Chengdu: "5MW Low Power Reactor Inherent Safety Research"; manuscript received 20 February 1992]

[Text] Abstract: Theoretical computations and reactor tests indicate that the 5MW low power reactor (5MW LPR) has rather good inherent safety. Under a loss of external power source accident reactor shutdown working conditions, it can rely on its own natural circulation and the heat storage capabilities without relying on any another safety facilities to remove residual heat from the core and guarantee the safety of the reactor.

Key terms: 5MW low power reactor, natural circulation, inherent safety, flow inversion.

I. Introduction

Due to historical reasons and modifications in the design program, the safety facilities of the 5MW LPR are relatively simple. It has no independent safety power source or safety-grade emergency cooling system, but theoretical analysis and a series of experiments in the reactor indicate that the 5MW LPR has rather good inherent safety capabilities. After a reactor shutdown due to a loss of external power source accident, the safety of the reactor can be guaranteed merely by relying on its own natural circulation and heat storage in the large 8.5 m x 2.2 m x 8.98 m water pool.

II. Steady-State Natural Circulation Capability Analysis

A. Extrapolation of the Natural Circulation Capabilities of the 5MW LPR Based On the Results of Tests in the HFETR

The core structure of the 5MW LPR is basically identical to the HFETR. They differ only slightly in the structure of their heat shielding. Water can flow through the heat shield of the HFETR but it cannot in the 5MW LPR. Experiments have already shown that the HFETR has rather substantial natural circulation capabilities (Table 1).

Table 1. Results of Test of Natural Circulation Capabilities for the First Heat at the HFETR (Core Loaded With 25 Boxes of Elements)

Reactor power, kW	Average power per box, kW	Average water temperature of element box inlet/outlet, °C	Average temperature of hot section (core)/cold section, °C	Temperature differential of cold/hot sections, °C
100	3.846	40.21/16.53	28.37/15.76	12.60
300	11.538	56.25/17.04	36.64/16.02	20.62
500	19.231	65.78/17.31	41.54/16.15	25.39
900	34.615	80.61/17.82	49.21/16.41	32.80
1200	46.150	96.48/18.34	57.41/16.66	40.75
1500	57.690	105.99/18.59	62.29/16.80	45.50

If we assume that the core structure of the low-power reactor is completely identical to the HFETR and if we conservatively stipulate that the mean temperature at the outlet of the 5MW LPR under a natural circulation state cannot be permitted to exceed 90°C, and if we assume that 20 percent of the power is generated in the cold section (beryllium reflecting layer, part of the heat shield, stainless steel blocks and aluminum filler blocks), we can derive a mean temperature differential of 25°C between the cold section and the hot section (core). We can determine from Table 1 that a temperature differential of 25°C between the cold and hot sections can form a natural circulation capability of 18.0 kW/box, so the reactor's total of 32 boxes of elements have a total natural circulation capability of 576 kW. As for the effects of differences in the structure of their heat shields on natural circulation capabilities, comprehensive debugging measurements from the HFETR and the calculated flow rate portions flowing through the various channels in the reactors (Tables 2 and 3) show that the calculated flow rate that flows through the heat shield is only 5.23 percent while the flow rate through the reflecting layers, control rods, and gaps in the core exceeds 60 percent.

Table 2. Flow Rates Through Each Channel in the HFETR After Comprehensive Debugging

Component	Number	Calculated value		Actual value	
		Flow rate, t/h	Percent	Flow rate, t/h	Percent
Fuel elements	44	2430	47.8	2332	45.8
Control rods containing elements	10	382	7.62	405	7.95

Table 2. Flow Rates Through Each Channel in the HFETR After Comprehensive Debugging (Continued)

Component	Number	Calculated value		Actual value	
		Flow rate, t/h	Percent	Flow rate, t/h	Percent
Control rods containing beryllium	8	112	2.20	144	2.33
Other	-	2162	42.2	2204	43.42
Total flow rate	-	5085	-	5085	-

Table 3. Results of Various Flow Channel Allocation Computations for First Heat at HFETR

Component	Number	Single item flow rate, t/h	Total flow rate	
			t/h	Percent
Fuel elements	25	-	1395.53	35.89
Control rods containing elements	4	39.22	158.08	4.07
Control rods containing beryllium	14	17.12	239.68	6.16
Beryllium reflecting layer	48	2.22	106.56	2.74
Cobalt target	2	4.23	8.46	0.22
Target 502	8	11.06	88.48	2.23
Internal heat shield	1	203.29	203.29	5.23
Gaps	-	-	1688.00	43.42
Total reactor flow rate	-	-	3888.0	-

Tables 2 and 3 show that under a natural circulation state, the flow channels in the HFETR heat shield serve as one part of the cold section channel and account for only 8.5 percent of the total cold section flow rate. Inside the 5MW LPR, this 8.5 percent flow rate will be allocated to other cold section channels. This means that under a situation of other conditions being identical, the cold section flow resistance in the 5MW LPR will be 18 percent greater than the HFETR. There is no harm is stating that it would be conservative to consider that due to the differences in their heat shields, the natural circulation capability in the 5MW LPR would be 18 percent less than the HFETR.

In addition, because of shielding and protection requirements, the 5MW LPR has 48 blocks whose filling material is stainless steel (the others are aluminum alloy), whereas the filling blocks in the HFETR core are all aluminum alloy. This means that at the same power, the heat generated in the 5MW LPR filling block region would be greater than in the HFETR. However, in the heat shield region, the heat shield material in the HFETR is also stainless steel, so it would also transfer a substantial amount of heat to the cold section liquid, but the 5MW LPR does not have this. The overall result of these two differences is that the heat generation component in the cold section of the 5MW LPR should be considered to be greater than in the HFETR, but this has already been included in the assumptions outlined above (20 percent of the power is generated in the beryllium block reflecting layer, aluminum filling blocks, and part of the heat shield). The reason is that in regular experimental reactors all of the non-fuel heat generation components account for only about 10 percent. To take a conservative view, we still consider 10 percent of indeterminate factors to be the difference between part of the filling block materials, which would have a negative impact on the natural circulation capabilities of the 5MW LPR.

After taking these revisions into consideration and adding several conservative margins, the natural circulation capability of the 5MW LPR reactor would be no less than 364.3 kW.

B. Theoretical analysis of the natural circulation capabilities of the 5MW LPR

Based on the actual structure of the 5MW LPR, theoretical computations indicate that if the coolant inlet temperature does not exceed 45°C and the heat channel outlet water temperature is lower than 90°C, the natural circulation flow rate in the reactor at this time would account for 5.21 percent of the rated flow rate and could remove 390.96 kW of heat.

III. Analysis of Residual Heat Removal Capabilities

Assuming that the reactor shuts down immediately when an accident causes the main pumps to cease operation but that the emergency cooling system does not go into operation, there could be complete reliance on natural circulation to remove the residual heat from the core into the reactor water pool.

Calculations indicate that 1.2 seconds after the main pumps cease operation, the reactor would rely on a "main flow rate low protection signal" for a protective reactor shutdown. At 44.5 seconds, core coolant flow inversion would begin and natural circulation would begin to be established. During the period of flow inversion, the heat channel outlet water temperature and element cladding surface temperature would reach their peak values at 104°C and 110°C, respectively. After natural circulation is established, the reactor would become increasingly safe.

As natural circulation in the core is established, the residual heat released from the core would continually be transferred from the core to the large water pool. The large water pool of the 5MW LPR contains over 150 tons of ≤ 45°C light water. In a situation in which the secondary cooling system ceases to function, it would take about 100 hours after the reactor is shut down for all of the residual heat that is released to heat the water in the pool that is 5 m deep at the top of the reactor from 45°C to a saturation temperature (not counting heat dissipation losses from the water pool).

This means that if the flow of the coolant in the 5MW LPR is interrupted, after the reactor shuts down because of the accident it could rely on its own natural circulation to remove the residual heat it releases into the large water pool outside of the reactor, and it could rely on its own enormous water volume and would not require cooling by the secondary cooling system to be able to absorb all of the residual heat released from the core. Thus, one can say that the 5MW LPR itself has very good inherent safety.

IV. Experimental Research in the 5MW LPR

To examine and confirm the inherent safety capabilities of the 5MW LPR reactor, a series of research experiments were conducted in the 5MW LPR reactor. The primary research projects were: measurement of main pump inertial flow rates, measurement of surplus power, power scale under forced circulation working conditions, steady-state natural circulation capability experiments at different powers, and flow inversion experiments at different powers.

The steady-state natural circulation capability experiments show that when the reactor is operating at a 350 kW steady-state power, the reactor is safe when relying entirely on natural circulation to cool the core and the heat channel outlet water temperature is only 79°C

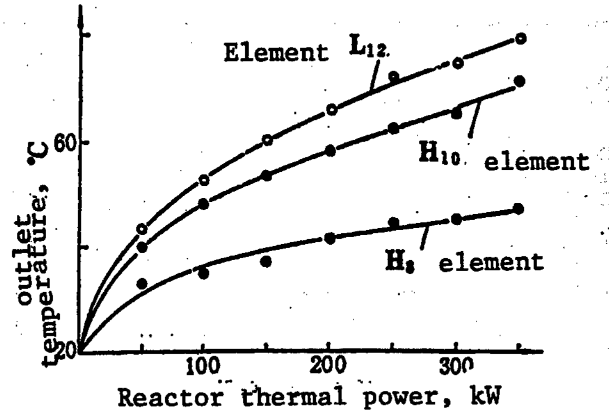

Figure 1. Curve of the Relationship Between Temperature at the Top End Outlet of Element Boxes and Reactor Thermal Power Under Natural Circulation Conditions

(Figure 1). This shows that the reactor has over 350 kW of natural circulation capability.

The results of the surplus power experiments (Figure 2) show that about 10 seconds after reactor shutdown, the reactor power would drop to 7 percent of the rated value, which is 350 kW. At this time, the inertial flow rate would still exceed the rated value by 20 percent, so cooling and removal of the residual heat would take far less than the 350 kW natural circulation capability.

The flow inversion experiments involved suddenly shutting down the operation of the main pumps when the reactor was operating at a relatively low power without shutting down the reactor and using thermocouples that had been previously installed at both ends of the hot elements to measure changes in the outlet temperature of the heat channels. The experiments were carried out at four powers, 100 kW, 150 kW, 200 kW, and 250 kW. The results show that flow inversion was safely achieved in all cases. The main results are illustrated in Figure 2 and Table 4.

The experiments can be diagnosed and confirmed by the following points. If while operating at the rated power of 5MW a shutdown of the main pumps occurs and the reactor immediately shuts down, relying on its own natural circulation cooling capability can completely remove the residual heat from the core and guarantee the safety of the reactor.

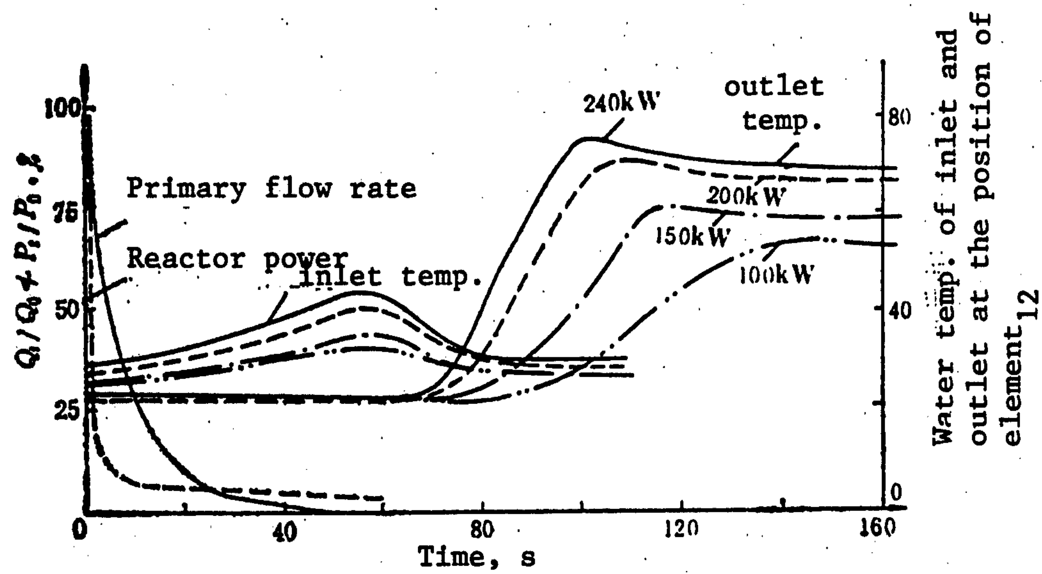

Figure 2. Results of Flow Inversion Experiments at Different Powers

Table 4. Main Results of Flow Inversion Experiments

Item			Reactor power, kW			
			100	150	200	240
Fuel region coolant temperature rise prior to pump shutdown, °C			0.29	0.43	0.53	0.69
Heat channel temperature prior to pump shutdown, °C	Upper end		22.0	22.0	22.0	23.0
	Lower end		25.5	25.5	27.5	29.0
Heat channel temperature during flow inversion process, °C	Upper end	Start	22.0	22.0	22.0	23.0
		Completion	54.0	59.5	69.0	73.0
	Lower end	Start	32.0	35.0	40.0	43.0
		Completion	26.5	26.5	28.5	30.0
Time of inversion process, s		Start	62.0	60.0	58.3	56.5
		Completion	147.0	116.0	108.8	102.5
Stable natural circulation heat channel temperature, °C	Upper end		51.0	57.0	64.5	66.5
	Lower end		26.0	26.0	27.5	30.0

A. Conducive to achievement of flow inversion

The situation when a main pump shutdown accident occurs at a power of 5MW is even more conducive to the achievement of flow inversion that under the experimental working conditions. Table 4 shows that the mean temperature rise of the coolant at the four experimental power levels was extremely small prior to pump shutdown. At a power of 240 kW, for example, the mean coolant temperature rise in the fuel region was less than 0.80°C and the mean coolant temperature rise in the heat channels was also only 6.0°C. This means that at the experimental power, the difference in the coolant density between the cold and hot sections, which is the motive force that forms natural circulation, was extremely small, so it would be difficult to achieve flow inversion to form natural circulation and would result in the appearance of a stagnation time of 1 to 5 seconds in the core's coolant, which is very unfavorable for heat transfer.

When operating at the rated power of 5MW, however, the mean coolant temperature rise in the core's fuel region exceeds 12°C and 22°C in the heat channels. After a reactor shutdown from a pump shutdown accident, although there may be some drop in the temperature rise during the time period from 5 to 20 seconds, it would still be higher than the value during the experimental working conditions (240 kW) (Figure 2 shows that the

power and flow rates during this period of time would still be 3 times to 1.25 times higher than the 240 kW experimental working conditions). After 20 seconds, as the inertial flow rates in the main pumps drop abruptly and decay power moderation declines, the temperature rise in the core coolant would gradually increase and before the inertial flow rate is zero, the coolant temperature rise in the fuel region would rise to and even surpass the level prior to the accident. At a relatively high hot section coolant temperature, this would inevitably create a rather large drive force that would aid in establishing natural circulation. This can be confirmed by the trends in variation of the flow inversion time with power in the examples given in Table 4. This means that it would not take 44.5 seconds after pump shutdown to achieve flow inversion as derived from analysis of the calculations above, and flow inversion would be achieved (assuming that the surplus power when flow inversion occurs is 240 kW) prior to the inversion time measured at the 240 kW experimental working condition.

B. Favorable factors in the flow inversion process

During the flow inversion process from a reactor shutdown due to a pump shutdown at 5MW, there would be even more favorable factors compared to the inversion process at the experimental working conditions:

1. A zero flow rate exists for only a short period of time or basically does not exist during the process of establishing natural circulation under pump shutdown and reactor shutdown working conditions at 5MW (except for a brief zero flow rate at the time of inversion). This undoubtedly aids in heat transfer and reduction of the temperature of the walls of the fuel elements because the heat transfer capabilities under a flowing state and stagnant flow state are substantially different at identical power densities. The heat transfer capability in the latter case is greatly reduced and the direct result may be an increase in the temperature of the walls of the elements. The longer the time of a zero flow rate, the more serious the outcome.

2. During the flow inversion process, the surplus power at a 5MW pump shutdown and reactor shutdown working condition is somewhat lower than at the experiment working condition of 240 kW. Figure 2 shows that the surplus power does not drop to 240 kW until 38 seconds after reactor shutdown while flow inversion occurs after 38 seconds, so it begins when the power is lower than 240 kW. It is very apparent that the lower the power at the time of flow inversion, the greater the safety.

C. Thermotechnical parameters will not attain the design limits

The thermotechnical parameters throughout the residual heat release and cooling process during a reactor shutdown due to a pump shutdown in the 5MW LPR will not attain the design limits. Table 4 shows that during the process of flow inversion at a constant power of 240 kW, after the heat channel coolant is heated for 46 seconds (starting at 56.5 seconds and ending at 102.5 seconds), it rises from 43°C to 73°C. In a 5MW pump shutdown and reactor shutdown working condition, because a relatively large drive head exists, premature flow inversion cannot be achieved, but it can shorten the process of establishing natural circulation (the time trends for flow inversion in the four experimental working conditions confirm this point). This means that the time that the heat channel water is heated during the inversion process is somewhat shorter. Moreover, the analysis above shows that the power level during the inversion process will also be somewhat lower than the 240 kW in the experimental working condition. Thus, at a 5MW reactor shutdown working condition, the temperature rise in the heat channel water during the flow inversion process will not exceed the 30°C attained at the 240 kW experimental working condition.

Based on the principles of establishing natural circulation (drive head equal to flow resistance) and the highest measured heat channel outlet water temperatures at the time of achieving flow inversion in the four types of experimental working conditions (32°c, 35°C, 40°C, and 45°C, respectively), it can be expected that the heat channel outlet water temperature during the process of achieving flow inversion at a 5MW reactor shutdown working condition will certainly not reach 82°C.

Since the initial highest water temperature is less than 82°C and the temperature rise during the inversion also will not reach 30°C, the highest water temperature during the inversion process will not attain the design limit of 112°C.

V. Conclusion

Based on the above analysis and experimental examination and confirmation, it is entirely possible to confirm that the 5MW LPR has rather good inherent safety. If a main pump shutdown accident occurs at the rated power of 5MW, immediately shutting down the reactor can guarantee its safety without requiring any other safety facilities by relying on its own natural circulation capabilities and the enormous heat absorption capacity of the large water pool.

On 17 January 1992, a main pump power outage reactor shutdown experiment was conducted in the 5MW LPR at the rated power of 5MW. The results of the test confirm that the above conclusion is entirely correct. The heat channel outlet water temperature during the experimental process was 59°C (corresponding to an inlet water temperature of 21°C) and the time of flow inversion was 65 seconds after the main power power outage.

Quality Assurance of 5MW Low Power Reactor

926B0125M Beijing HE DONGLI GONGCHENG [NUCLEAR POWER ENGINEERING] in Chinese Vol 13, No 4, 10 Aug 92 pp 87-90

[Article by Liu Fenglin [0491 7685 2651] of the China Nuclear Power Research and Design Academy, Chengdu: "5MW Low Power Reactor Quality Assurance"; manuscript received 20 February 1992]

[Text] Abstract: This article provides a brief introduction to the quality assurance system in the design, construction, debugging and operation of the 5MW LPR.

Key terms: 5MW low power reactor, quality assurance system, quality assurance program, record system.

I. Introduction

The 5MW low power reactor (abbreviated below as the 5MW LPR) is a research- type reactor. While its operation parameters are relatively low, it is still a technically complex project. For this reason, it must be managed scientifically. The quality assurance system described in this article is an effective measure for this type of scientific management.

China nuclear safety regulation HAF0400, "Nuclear Power Plant Quality Assurance Stipulations" provide this definition of quality assurance: "All planned and organized activities to establish confidence that a certain thing or a certain set of equipment will truly operate satisfactorily when used in the future". To attain the quality and the stipulated technical performance demanded of things and services by nuclear safety regulations, we established and perfected a quality assurance system in 1989. Construction of the 5MW LPR was carried out in a planned and organized manner according to quality assurance procedures.

II. Quality Assurance Organization Structure for the 5MW LPR

The China Nuclear Power Research and Design Academy is the owner of the 5MW LPR as well as the unit that designed, built, and manages it. The quality assurance organization mechanisms for this reactor are illustrated in Figure 1.

The academy director has the scope of his duties demarcated in the "Policy Statement" in the "Low-Power Reactor Quality Assurance Program": the academy director is responsible for the effectiveness of the program and the quality of overall construction; quality assurance departments are responsible for supervising implementation of the program; the other relevant work personnel and management personnel bear responsibility for the quality of their own tasks. Figure 1 also stipulates the channels that connect each organization. This enables mutual coordination and matching up of all items of work.

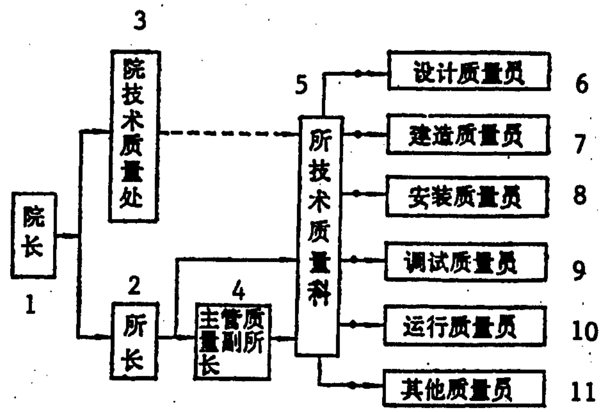

图1 5MW LPR质保组织机构

——指令关系; -·-监督关系; ———指导关系.
a b c

Figure 1. Organizational Structure of Quality Assurance for the 5MW LPR

Key: 1. Academy director; 2. Institute director; 3. Academy Technical Quality Department; 4. Assistant institute director in charge of quality assurance; 5. Institute Technical Quality Office; 6. Design quality personnel; 7. Construction quality personnel; 8. Installation quality personnel; 9. Debugging quality personnel; 10. Operational quality personnel; 11. Other quality personnel; a. Command relationship; b. Supervisory relationship; c. Guidance relationship

III. Quality Assurance Document System and Implementation

According to China's nuclear safety regulation HAF1000 "Research Reactor and Critical Device Operational Safety Stipulations" and HAF0400 "Nuclear Power Plant Quality Assurance Safety Stipulations", the academy director is authorized to organize the relevant personnel responsible for technical quality departments in the China Nuclear Power Research and Design Academy to formulate the "Low-Power Reactor Quality Assurance Program" (abbreviated below as the quality assurance program). This program is composed of four sub-programs in four phases (design, construction, debugging, and operation) and is the legal document for quality management during construction of the 5MW LPR. Based on the requirements in the quality assurance program, various types of associated implementation procedures were also formulated and concrete operational steps were also formulated as needed on the basis of the implementation procedures. The state's laws, regulations, and guiding products concerning nuclear safety are the first level in quality assurance for construction of the low-power reactor, the low-power reactor quality assurance program is the second level, the various implementation procedures are the third level, and the operational steps and operation cards are the fourth

level. The documents at these four levels constitute a complete quality assurance system.

The four sub-programs in the low-power reactor quality assurance program have these common characteristics:

1. They all belong to the second level of the quality assurance program and have the same legal effect in low-power reactor construction and operation;

2. In terms of format and content, they make the corresponding stipulations regarding standards, regulations, baselines, organizational forms, personnel qualifications, responsibilities and rights, examination, supervision, inspection and acceptance, and so on. They also set forth the corresponding requirements for the form, interface, transmission, modification, recording, management, and so on for all types of documents;

3. The China Nuclear Power Research and Design Academy Quality Department carries out internal supervision on the basis of the stipulations in HAF0409 "Nuclear Power Plant Quality Assurance Inspection" and the academy's "Quality Assurance Inspection Management Procedures".

I will now briefly describe the work in each phase of the low-power reactor quality assurance program.

A. Quality assurance in the design phase

This phase of the quality assurance program sets forth the quality requirements for the design of things with safety importance. Because there were still no clear quality assurance requirements in the design phase for the 5MW LPR, only retrospective inspection could be done in this phase. All of the design inputs in the design process (such as design standards, etc.) were clearly embodied in technical documents, blueprints, procedures, instructions, or manuals. Every effort was made to do design inspections and checking (alternative) computations and other types of design examination and acceptance for all designs that played important roles in safety (physical, thermotechnical, radiation protection, earthquake resistance, etc.).

B. Quality assurance in the construction phase

Requirements were set forth based on HAF0404 "Nuclear Power Plant Construction Period Quality Assurance" and in conjunction with concrete conditions for construction activities (including civil engineering construction, item purchasing, equipment processing and installation, etc.) and materials inspection and acceptance, technical examination and acceptance, while strength, specific gravity, and other experiments were carried out during construction for things like grouting of the concrete shield. Comprehensive records were established for the shielding project. Comprehensive item inspections were carried out prior to installation of equipment and systems. For example, 100 percent flaw detection and integrity, cleanliness, and other inspections were made of the piping in the main cooling system in the reactor pool and satisfactory corrections were made for all items that did not conform (the connecting bolts for the flanges of the main heat exchangers were too short, etc.).

C. Quality assurance in the debugging phase

Based on the projects and objectives in HAF0405 "Nuclear Power Plant Debugging and Operational Period Quality Assurance" and HAF0304 "Nuclear Power Plant Debugging Procedures", and in conjunction with the basic situation at the 5MW LPR, we compiled that "Low-Power Reactor Debugging and Testing Program" (abbreviated below as the debugging program).

The debugging program divided the overall debugging process for the 5MW LPR into three stages. Stage A was the pre-operational phase (also divided into single system debugging and comprehensive debugging). Stage B was loading, initial criticality, and low-power testing. Stage C was increased power testing. The main tests in this phase involved 15 important tests including reactor residual heat removal capability tests, power scaling tests, equilibrium xenon poisoning and iodine trapping tests, control and protection system readjustment and setting tests, radiation and environmental monitoring, 72-hour full-power operation, and so on.

In addition, based on the low-power reactor debugging quality assurance program, stipulations were prepared based on appropriate procedures and detailed principles for all activities that affect quality, and over 60 technical and administrative procedures for debugging were compiled, as were quantitative or qualitative examination and acceptance standards corresponding to the requirements in the procedures and detailed principles to confirm that all types of work that affected quality were completed satisfactorily.

To ensure the effectiveness of the debugging procedures, all of the procedures underwent three-level assessments for proofreading, examination and verification, and approval prior to their utilization, and they were approved by the National Nuclear Safety Administration. Besides stipulation the debugging content, methods, steps, and examination and acceptance methods for each procedure, the prerequisite conditions, limits, personnel qualifications, safety attention points, and so on were also clarified.

The Debugging Leadership Group was responsible for on-site debugging work. The Debugging Leadership Group's director was an administrative assistant institute director (a senior engineer) and its deputy group director was an assistant chief engineer (the person with technical responsibility). The leadership group was composed of personnel with practical experience and specialized technical knowledge from the institute's Science and Technology Office, design, construction, operation, and other units.

Safety and protection departments carried out safety monitoring of the debugging activities. Quality assurance departments supervised implementation of the debugging program and implemented on-site monitoring and control and inspection tours of important debugging activities. After the debugging activities were completed, examination and acceptance was carried out according to the examination and acceptance standards. After being examined and accepted as conforming to specifications, personnel responsible for debugging and quality assurance personnel signed the permits. In summary, all of the debugging work for the 5MW LPR was carried out safely and effectively under conditions of organization, foundation, and supervision. During the entire debugging period, a total of several 10 items of debugging work were completed (including 10 important debugging projects that were inspected by the National Nuclear Safety Administration) and the associated debugging summarization reports were written.

D. Quality assurance in the operation phase

In accordance with China's nuclear safety law HAF1000 "Research Reactor and Critical Device Operational Safety Stipulations", HAF0404 "Nuclear Power Plant Construction Period Quality Assurance", and HAF0405 "Nuclear Power Plant Debugging and Operational Period Quality Assurance", and in conjunction with actual conditions at this reactor, technical and administrative regulations were compiled for the operation phase. Their main content includes safety regulations, operation limits and conditions, operation regulations, inspection and repair regulations and management systems, system startup and shutdown operation cards (detailed principles), reactor opening or reloading inspection, and so on to enable reactor startup, shutdown and repair, maintenance, and so on to be under the control of quality assurance procedures and to provide rules to follow.

To make work during all phases of design, construction, installation, and debugging at the 5MW LPR conform to the stipulated quality requirements, besides formulating a complete set of document systems, specific requirements were also proposed for personnel qualifications, operating skills, quality consciousness, and so on. Among them, special attention was given to the qualifications, skill, knowledge, enthusiasm, and sense of responsibility of operating personnel. Actual operation was implemented in strict accordance with stipulations in the Nuclear Industry Corporation's "Reactor Operating Personnel Licensing System". Only personnel holding licenses could work at the corresponding posts and those without licenses were prohibited from going to posts. Those holding lower-grade licenses cannot work at higher-level posts. Full-power operation for nearly 70 shifts and 154 hours confirmed that the operational safety of this reactor was satisfactory.

In addition, clear requirements were set forth for dealing with emergencies, analyzing accidents, operational summarization and evaluation, and so on to perfect the quality assurance requirements for this reactor during its entire operational lifespan.

IV. Quality Assurance Records

Records are objective evidence regarding whether or not things or services conform to quality assurance requirements. If there are no records or the records are incomplete, it is difficult to evaluate the quality of this item of work. At the same time, these records are the only basis of whether or not there has been effective adherence to the quality assurance program and procedures, and they can provide important basic evidentiary data for improving this item of work or perfecting services. A corresponding quality assurance record system was established for the 5MW LPR and the on-site leadership group established quality assurance record management posts guided by specialized quality assurance personnel (the office quality assurance leadership group during the period of reactor operation) to assume responsibility for supervision, collection, processing, and storage of all categories of technical data (over 350 types) and quality assurance records so that there was data that could be inspected for all items of work at the 5MW LPR.

V. Conclusion

Effective management is the essence of quality assurance work. Quality assurance work at the 5MW LPR was subject to restriction by the relevant nuclear safety regulations and supervision by quality assurance departments. The establishment of the quality assurance system meant that "there are people responsible for every matter, there are regulations that can be followed in work, and there are data that can be examined for the results" for every phase at the 5MW LPR, which ensured smooth construction of the 5MW LPR. The single-system debugging, comprehensive debugging, nuclear debugging, 72-hour trial operation and 82-hour full-power trial production operation of this reactor confirmed that its design and construction both attained the expected quality requirements.

Environmental Impact of 5MW Low Power Reactor

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[Article by Xiong Dewu [3574 1795 3814] of the China Nuclear Power Research and Design Academy, Chengdu: "5MW Low Power Reactor Environmental Impact"; manuscript received 28 January 1992]

[Text] Abstract: This article assesses the environmental impact of radioactive effluent from the 5MW low power reactor (5MW LPR). Data from a population and dietary survey of the area around the plant site along with

computing models and parameters[1] were used to estimate that during normal operation of this reactor, the maximum individual effective dose equivalent for an area 1 km from the boundary of the plant site was 8.89×10^{-8} Sv x a^{-1} and a collective effective dose equivalent within a range of 80 km was 7.17×10^{-4} person x Sv x a^{-1}. The survey data indicate that the environmental impact of this reactor on this region during normal operation and hypothetical accidents is acceptable.

Key terms: 5MW low power reactor, dose equivalent, environmental quality.

I. Introduction

To assess the environmental impact of the 5MW LPR during its period of operation, given the need to assess the radiation and environmental quality during the first loading phase at the reactor, survey data composed of local climate, hydrology, natural resources, population distribution, and diet as well as computing models and parameters[1] were used to estimate that the maximum individual effective dose equivalent 1 km from the boundary of the plant site during normal operation of the low-power reactor was 8.89×10^{-8} Sv x a^{-1}, and most of this dose would come from ingestion of ^{137}Cs. The collective effective dose equivalent with a range of 80 km would be 7.17×10^{-4} person x Sv x a^{-1}. The environmental impact of the reactor under normal operation working conditions and accident working conditions was assessed.

II. Plant Site Location and Environment

The 5MW LPR is located in Jiajiang County, Leshan City, Sichuan Province at east longitude 103°29' and north latitude 29°26', 36.5 kilometers from Leshan City, 10.5 km from Jiajiang County, and 1.5 km from the eastern side of the Qingyi Jiang. The area is surrounded by mountains on four sides and the Nan'an Jiang flows circuitously from the plant area and into the Qingyi Jiang.

The area within 5 km of the plant region has a hilly terrain that slopes gently from southwest to northeast. To the northeast is the Qingyi Jiang dam site section, and the topography there is level and open. It is about 440 m above sealevel.

The region where the plant site is located has a moderately subtropical moist monsoon climate with 171 days of precipitation and an average yearly rainfall of 1,405.7 mm. The plant region mainly has category D weather, accounting for 68.09 percent. The mean annual wind speed is 2.06 m/s and the primary wind direction is SW.

Centered on this reactor, the total population within a radius of 80 km in 1990 was 7.68 million (extrapolated from 1982 population S&T statistical data).

A survey of the agricultural and animal husbandry production situation around the plant site showed that the primary agricultural products of this region are paddy rice, rape, wheat, tea, vegetables, and fruits. Animal husbandry mainly involves raising in household pens and the main livestock is hogs. The annual amounts of staple and non-staple foods consumed by the residents of this regions are: 254 kg x a^{-1} in grains, 203 kg x a^{-1} in vegetables, and 12 kg x a^{-1} in pork.

III. Source Items

To assess the environmental impact of radioactive effluents when the 5MW LPR was placed into normal operation during 1990, we took into consideration the extreme design working condition of this reactor. One of the main conditions assumed in this extreme working condition was an assumption of an 0.5 percent damage rate to the fuel elements in the reactor. In the assessment, it was assumed that after 99.5 percent of the radioactive effluent had been filtered by the ventilation central filtering system and was discharged into the atmosphere from a 125 m smokestack, there would be only a small amount of radioactive effluent that would leak from doors and windows. The amounts of radioactive gaseous effluent released are listed in Table 1.

Table 1. Amounts of Primary Nuclides Released in Gaseous Effluent Under Extreme Design Working Conditions, Bq

Nuclide	^{41}Ar	^{85}Kr	^{85m}Kr	^{87}Kr	^{88}Kr	^{131m}Xe	^{133m}Xe	^{133}Xe	^{135}Xe
Released at high elevation	1.73×10^{11}	2.31×10^{8}	7.36×10^{9}	1.38×10^{10}	2.15×10^{10}	2.68×10^{8}	9.15×10^{8}	4.97×10^{10}	5.08×10^{10}
Released at ground surface	8.68×10^{8}	1.16×10^{6}	3.70×10^{7}	6.93×10^{7}	1.08×10^{8}	1.35×10^{6}	4.60×10^{6}	2.50×10^{8}	2.55×10^{8}
Nuclide	^{135m}Xe	^{131}I	^{132}I	^{133}I	^{134}I	^{135}I	^{137}Cs	^{60}Co	^{3}H
Released at high elevation	1.37×10^{10}	2.37×10^{7}	3.50×10^{7}	3.21×10^{7}	6.02×10^{7}	4.70×10^{7}	6.04×10^{6}	5.39×10^{5}	1.70×10^{9}
Released at ground surface	6.90×10^{7}	1.19×10^{5}	1.76×10^{5}	1.62×10^{5}	3.03×10^{5}	2.36×10^{5}	3.03×10^{4}	2.71×10^{3}	8.55×10^{6}

All of the various types of waste liquids generated by the 5MW LPR are collected in the waste water workshop where they undergo evaporation treatment. When the condensed liquid attains discharge standards permitted

by the state, it is drained into the Qingyi Jiang. The total activity of the ^{3}H released into the environment is 1.98 $\times 10^{10}$ Bq x a^{-1} and the total activity of the ^{60}Co is 1.48 $\times 10^{7}$ Bq x a^{-1}.

See Table 2 for the released source items under the maximum hypothetical accident. This accident assumes an accident working condition in which one box of nuclear fuel elements is melted by overheating. Under this type of working condition, 25 percent of the total amount of equilibrium radioactive iodine in the core and 100 percent of the accumulated radioactive inert gases leak to the outside.

Table 2. Released Source Items Under Maximum Hypothetical Accident

Nuclide	^{85}Kr	^{85m}Kr	^{87}Kr	^{88}Kr	^{131m}Xe	^{133m}Xe	^{133}Xe	^{135}Xe	^{135m}Xe	^{131}I	^{132}I	^{133}I	^{134}I	^{135}I
Amount released	1.45×10^{12}	4.63×10^{13}	8.65×10^{13}	1.35×10^{14}	1.68×10^{12}	5.76×10^{12}	3.12×10^{14}	3.20×10^{14}	8.62×10^{13}	1.66×10^{11}	2.45×10^{11}	2.25×10^{11}	4.21×10^{11}	3.29×10^{11}

IV. Computation Methods

The dose estimates are based on the dosimetry models and related parameters for evaluating the environmental quality of radiation in reference [1] and on the accidental release into the atmosphere dispersion computation model in the United States NRC management guiding principle 1.4[2] and were done on a computer. To make the results of the calculations conform more to reality, every effort was made to use the relevant parameters obtained from actual surveys and on-site experiments, such as atmospheric stability and its frequency, discharge outlet wind rose, wind direction and wind speed combined frequency, river hydrology parameters, resident population distribution and diet, and so on.

V. Dose Estimate Results

A. Annual Effective Dose Equivalent for Individual Residents

The maximum individual effective dose equivalent among residents at the boundary of the plant site from air-borne releases is 8.89×10^{-8} Sv x a^{-1} and the individuals receiving the maximum irradiation dose would be members of a group of infants at a position 1 km to the northeast.

Table 3 lists the contributions of each type of gaseous radioactive nuclide released via each type of irradiation route on the maximum individual annual effective dose. In Table 3, ^{137}Cs makes the greatest contribution, about 78 percent, followed by the contributions of ^{41}Ar, ^{60}Co, ^{131}I, and ^{88}Kr. In terms of irradiation routes, the ingestion route makes the greatest contribution, about 57 percent, followed by external irradiation from deposits on the ground surface, which accounts for 30 percent, and by external irradiation from immersion in smoke and clouds, which accounts for 13 percent, whereas the contribution from inhaling is very small and can be ignored. It is apparent that the irradiation received by residents does not come primarily from direct external irradiation from smoke and clouds and internal irradiation from inhalation, but instead comes from internal irradiation from eating and external irradiation from radioactive deposits on the ground surface.

The maximum individual effective dose equivalent from liquid effluent for residents downstream from the discharge outlet on the Qingyi Jiang is 4.30×10^{-10} Sv x a^{-1} and the maximum irradiation dose is among members of a group of adults 1.5 km to the east-southeast. ^{60}Co makes the greatest contribution to the dose. In terms of irradiation routes, the main ones are external irradiation from riverbank deposits, followed by internal irradiation through the ingestion route. The results show that the contribution of liquid effluent to the dose of residents is smaller than that from gaseous effluent.

Table 3. Contributions of Each Nuclide and Each Irradiation Route at Each Release Point to the Maximum Individual Annual Effective Dose Equivalent*, Sv x a^{-1}

Irradiation route	External radiation via clouds and mist immersion		External radiation deposited on Earth's surface		Inhaled		Ingested		Total
Nuclide/Release point	1	2	1	2	1	2	1	2	
^{41}Ar	8.28×10^{-9}	1.67×10^{-10}	-	-	-	-	-	-	8.45×10^{-9}
^{85}Kr	2.10×10^{-14}	4.06×10^{-16}	-	-	-	-	-	-	2.14×10^{-14}
^{85m}Kr	4.85×10^{-11}	9.41×10^{-13}	-	-	-	-	-	-	4.94×10^{-11}
^{87}Kr	4.60×10^{-10}	8.91×10^{-12}	-	-	-	-	-	-	-4.69×10^{-10}

Table 3. Contributions of Each Nuclide and Each Irradiation Route at Each Release Point to the Maximum Individual Annual Effective Dose Equivalent*, Sv x a^{-1} (Continued)

Irradiation route	External radiation via clouds and mist immersion		External radiation deposited on Earth's surface		Inhaled		Ingested		Total
^{88}Kr	1.78×10^{-9}	3.46×10^{-11}	-	-	-	-	-	-	1.82×10^{-9}
^{131m}Xe	1.38×10^{-13}	2.67×10^{-15}	-	-	-	-	-	-	1.41×10^{-13}
^{133m}Xe	1.30×10^{-12}	2.52×10^{-14}	-	-	-	-	-	-	1.33×10^{-12}
^{133}Xe	8.23×10^{-11}	1.59×10^{-12}	-	-	-	-	-	-	8.39×10^{-11}
^{135}Xe	3.38×10^{-10}	6.56×10^{-12}	-	-	-	-	-	-	3.45×10^{-10}
^{135m}Xe	2.41×10^{-10}	4.67×10^{-12}	-	-	-	-	-	-	2.46×10^{-10}
^{131}I	3.64×10^{-13}	7.05×10^{-15}	1.48×10^{-10}	7.02×10^{-12}	6.73×10^{-11}	1.31×10^{-12}	3.39×10^{-9}	1.60×10^{-10}	3.78×10^{-9}
^{132}I	6.22×10^{-13}	1.21×10^{-14}	1.59×10^{-11}	7.54×10^{-13}	3.55×10^{-13}	6.88×10^{-15}	4.78×10^{-13}	2.25×10^{-14}	1.93×10^{-11}
^{133}I	6.98×10^{-13}	1.35×10^{-14}	2.90×10^{-11}	1.37×10^{-12}	1.45×10^{-12}	2.81×10^{-14}	1.46×10^{-11}	6.88×10^{-13}	4.78×10^{-11}
^{134}I	1.01×10^{-12}	1.96×10^{-14}	9.63×10^{-12}	4.55×10^{-13}	3.21×10^{-13}	6.23×10^{-15}	2.71×10^{-13}	1.28×10^{-14}	1.17×10^{-11}
^{135}I	3.49×10^{-12}	6.77×10^{-14}	4.39×10^{-11}	2.08×10^{-12}	1.05×10^{-12}	2.03×10^{-14}	5.22×10^{-12}	2.47×10^{-13}	5.61×10^{-11}
^{60}Co	5.31×10^{-14}	1.03×10^{-15}	3.80×10^{-9}	1.79×10^{-10}	3.82×10^{-12}	7.40×10^{-14}	1.44×10^{-10}	6.78×10^{-12}	4.13×10^{-9}
^{137}Cs	1.40×10^{-13}	2.72×10^{-15}	2.11×10^{-8}	9.95×10^{-10}	1.11×10^{-11}	2.15×10^{-13}	4.51×10^{-8}	2.13×10^{-9}	6.93×10^{-8}
^{3}H	-	-	-	-	3.05×10^{-12}	5.91×10^{-14}	9.72×10^{-12}	4.59×10^{-13}	1.33×10^{-11}
Total	1.13×10^{-8}	2.24×10^{-10}	2.51×10^{-8}	1.19×10^{-9}	8.85×10^{-11}	1.72×10^{-12}	4.87×10^{-8}	2.30×10^{-9}	8.89×10^{-8}

*Individuals receiving greatest dose are located at a site 1 km to the NE; **1 means release at high elevation, 2 means release at ground surface.

B. Annual collective effective dose equivalent

Table 4 lists the contributions to the collective effective dose equivalent from different release modes and different irradiation routes. Table 4 shows that the annual effective dose equivalent within a range of 80 km from air-borne releases is 6.65×10^{-4} person x Sv x a^{-1}. Ingestion and external irradiation from deposits are the primary irradiation routes. The nuclide making the greatest contribution to the collective dose equivalent is ^{137}Cs.

The annual effective dose equivalent within a range of 80 km downstream from the discharge outlet from liquid-borne releases is 5.15×10^{-5} person x Sv x a^{-1}. The irradiation route making the greatest contribution is internal irradiation from ingestion. The nuclide making the greatest contribution is ^{60}Co.

Table 4. Contributions to the Annual Collective Effective Dose Equivalent From Different Release Modes and Via Different Irradiation Routes

Irradiation route	External irradiation from immersion	Internal irradiation from inhaling	Internal irradiation from ingestion	External irradiation from deposits on the ground surface	Total
Air-borne releases	7.17×10^{-5}	5.92×10^{-7}	3.50×10^{-4}	2.37×10^{-4}	6.65×10^{-4}
Liquid-borne releases	3.65×10^{-10}	-	5.15×10^{-5}	1.20×10^{-8}	5.15×10^{-5}

C. Accident doses

Tables 5 and 6 list the possible effective dose equivalents and thyroid gland dose equivalents from irradiation for individuals 1 km from the boundary of the plant site during the largest hypothetical accident. Those receiving the greatest irradiation dose would be the members of a

group of infants 1 km to the northeast. Their individual effective dose equivalent would be 4.91 x 10^{-5} Sv and their thyroid gland dose equivalent would be 9.2 x 10^{-4} Sv. The primary irradiation route would be internal irradiation arising from ingestion and the primary nuclide would be ^{131}I.

During the largest hypothetical accident, the collective effective dose equivalent for residents within a range of 80 km would be 1.47 x 10^{-1} person x Sv x a^{-1}. External irradiation from immersion would make the greatest contribution to the collective dose.

Table 5. Maximum Individual Effective Dose Equivalent at the Boundary During the Largest Hypothetical Accident

Age group	External irradiation from immersion in smoke and clouds	External irradiation from deposits on the ground surface	Internal irradiation from inhaling	Internal irradiation from ingestion	Total
Infants	2.18 x 10^{-5}	1.70 x 10^{-6}	9.43 x 10^{-7}	2.47 x 10^{-5}	4.91 x 10^{-5}
Children	2.18 x 10^{-5}	1.70 x 10^{-6}	1.22 x 10^{-6}	1.91 x 10^{-5}	4.39 x 10^{-5}
Adults	2.18 x 10^{-5}	1.70 x 10^{-6}	6.34 x 10^{-7}	7.69 x 10^{-6}	3.19 x 10^{-5}

Table 6. Thyroid Gland Dose Equivalents for Individuals at the Boundary During the Largest Hypothetical Accident

Item	Age group		
	Infants	Children	Adults
Thyroid gland dose equivalent	9.20 x 10^{-4}	7.51 x 10^{-4}	3.05 x 10^{-4}
Location of concern	1 km to the northeast		

VI. Conclusion and Discussion

The maximum individual effective dose equivalent for residents from radioactive releases during normal operation of the 5MW LPR is 8.89 x 10^{-8} Sv x a^{-1}. This value is far below the estimated global per capita annual effective dose rate value of 2 mSv normal background generated by regional natural radiation sources provided in the UNSCEAR 1982 Report[3], and it is lower than the management limit allocated for this reactor of 0.05 Sv x a^{-1}.

1. The key population group is a group of infants 1 km to the northeast. The key irradiation route is internal irradiation from ingestion, followed by external irradiation from ground deposits. The key nuclide is ^{137}Cs, while ^{41}Ar, ^{60}Co, ^{131}I, and ^{88}Kr also made substantial contributions to the maximum individual dose equivalent. The release mode with the greatest environmental impact is air-borne releases. It is very important that every effort be made to rationally improve radioactive aerosol particles and iodine purification capabilities to reduce the irradiation dose to the public.

2. Under normal operation conditions, the annual collective effective dose equivalent within a range of 80 km would be 7.17 x 10^{-4} person x Sv x a^{-1}, which is far below the annual collective effective dose equivalent from natural radiation sources of 1.4 x 10^4 person x Sv x a^{-1}. Table 7 shows that the nuclide making the greatest contribution to the collective dose equivalent is ^{137}Cs and that the key irradiation route is internal irradiation from ingestion, followed by external irradiation from ground surface deposits.

Table 7. Maximum Individual Effective Dose Equivalent and Collective Effective Dose Equivalent Within a Range of 80 km

Release mode	Normal		Accident
Release route	Air-borne	Liquid-borne	Air-borne
Maximum individual effective dose equivalent, Sv	8.89 x 10^{-8}	4.30 x 10^{-10}	4.91 x 10^{-5}
Collective effective dose equivalent, person/Sv	6.65 x 10^{-4}	5.15 x 10^{-5}	1.47 x 10^{-1}
	7.17 x 10^{-4}		
Key nuclide	^{137}Cs		^{131}I
Key route	Internal irradiation from ingestion		Internal irradiation from ingestion
Key population group	Group of infants at a site 1 km to the northeast		

3. If the largest hypothetical accident occurs at this reactor, the effective dose equivalent received by individuals 1 km from the boundary during the accident would be 4.91 x 10^{-5} Sv, which is only 0.98 percent of the 5 mSv control limit for individuals of the public during each major accident in the state standard (GB6249-86).

The thyroid gland dose equivalent would be 9.20×10^{-4} Sv, which is only 1.84 percent of the 50 mSv control limit for the dose equivalent to the thyroid gland among individuals of the public during each accident in the state standard (GB6249-86).

4. During the entire period of the accident, the collective effective dose equivalent that could be generated within a range of 80 km is 1.47×10^{-1} person x Sv, which is lower than the 2×10^{4} limit for the collective effective dose equivalent received by population groups within a radius of 80 km under the largest imaginable accident conditions stipulated in the state standard (GB6249-86).

5. From the estimated environmental impact based on the HFETR, 5MW LPR, element production, and other nuclear facilities and adding the public dose creased by the operation of this nuclear facility, the collective effective dose equivalent received by population groups within a range of 80 km is 6.28×10^{-2} person x Sv x a^{-1} and the maximum individual effective dose equivalent is 4.70×10^{-6} Sv x a^{-1}, which is just 1.9 percent of the dose control limit of 0.25 mSv for individuals among the public stipulated in the state standard (GB6249-86).

The above data show that the environmental impact of the 5MW LPR on this region during normal operation and hypothetical accidents is acceptable.

References

[1] Pan Ziqiang [3382 5261 1730], He Gongye Sanshi Nian Fushe Huanjing Zhiliang Pingjia [An Assessment of Environmental Quality from 30 Years of Radiation from the Nuclear Industry], Atomic Energy Press, 1990, Chapter 2.

[2] U. S. NRC, Regulatory Guide, 1.4, 1974.

[3] UNSCEAR, 1982 Report, Annex C.F., 1982.